EXHIBIT A

original

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	)	**Docket No. 50-247**
License Renewal Application	)	

DECLARATION OF SUSAN H. SHAPIRO, Esq.

My name is Susan H. Shapiro I live at 36 Horne Tooke Road, Palisades NY 10964, less than 17 miles from Indian Point. I am President of Friends United for Sustainable Energy, USA, Inc (FUSE), a steering committee member of the Indian Point Safe Energy Coalition, and a board member of Hudson River Sloop Clearwater.

I am a life long resident of Rockland County, as is my father and grandfather. I enjoyed walking and cycling along the banks of the Hudson with my family. I am a mother of two children ages 8 and 10.

On 9/11 I first became concerned about the threat of terrorism to Indian Point, as the 9/11 hijackers flew directly over the plant and in fact, had plans to attack before they decided instead to attack the World Trade Center.

As I learned more about the operation of Indian Point, my concern about a terrorist attack, have been dwarfed by the seemingly endless operating problems and leaks at the aging facility, which the current owner, Entergy bought on or about 9/11.

At one of the first annual assessment meetings I attended in Buchanan, NY I was shocked to hear about the amount of repairs required When I expressed my concern an Entergy employee said "it's like an old car, we just keep patching it and it keeps on running". This was my introduction into the lack adequate aging management at the plants.

Since that time, there have been a series of chronic problems with the plant. In 2005 leaks of tritium where discovered accidently near spent fuel pool #2, further investigation uncovered large amounts of Strontium 90 apparently leaking from spent fuel #1. However, to date, the exact location, size, duration and methods of stopping and remediating said leaks remains unknown.

Other leaks seems to be sprouting up, and are being discovered only by accident, instead of through proper and thorough investigation.

I cannot understand how the NRC can possibly justify issuing Entergy a new superceding license to an additional 20 years, when the plant has clearly outlived its ability to be run without jeopardizing public health and safety, and the integrity of the environment.

The NRC's overly close relationship with the NEI, the nuclear industry's lobby group, became apparent to me, when during a conference in Washington, D.C., during Katrina, the NEI introduced a white paper that reduced the evacuation area guidance from the 10 mile radius, to a 2 mile wedge. NRC quickly rubberstamped favoring protection of the financial profits of the nuclear industry to those of public health and safety, as required by it's organizing mandate.

Indian Point is unique, as it is the only plant located in the middle of 21 million residents, 24 miles from New York City, 3 miles from West Point Military Academy, is leaking Strontium 90, tritium and cesium into the groundwater and Hudson River, and does not have an adequate, workable or fixable evacuation plan, an

Our elected officials, Federal, State and Local, and thousands of Hudson Valley residents have called for Indian Point closure and for an Independent Safety Assessment prior to consideration for relicensing. In fact, even though the NRC refused to require backup power for an emergency siren system, a Federal law was passed that did require such a system be installed and operable months ago. To date Entergy has been unable to properly install the required siren system.

I am aware that the plant is currently leaking dangerous radioactive contaminants from the plant into the ground around the plant, as well as the In the event the river continues to be contaminated from releases from Indian Point, my enjoyment of the river for recreation and exercise will be directly affected. In addition, if Rockland County needs to start using the river for our public water supply my health and the health of my children may be adversely affected.

Indian Point needs to be shutdown, I understand the law requires the site to cleaned up to the condition it was in prior to the plant being built. It appears that the law is being broken. For example in the case of Unit 1, which was shut down over 30 years ago, its spent fuel pool is currently leaking Strontium 90, tritium and cesium into the river. The river is continuing to be polluted as a result of the inaction of the owners and regulators.

If this was any other kind of business, such as a gas station, the government authorities would shut it down and make the owners remediate the underground leaks immediately.

Today, Indian Point could not be sited where it is located in the most densely populated region of the country, on a earthquake fault, and along with the inadequate aging management of the plant, the NRC cannot issue a new superceding license to the operator for another 20 years. In fact the plant should be closed immediately and the cite decommissioned.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 12 day of September, 2007, at Spring Valley, NY.

Susan H. Shapiro, Esq.

Sworn to before me this
12th day of September, 2007.

Patricia E. French

PATRICIA E. FRENCH
Notary Public, State of New York
No. 01FR5041486
Qualified in Rockland County
Commission Expires ~~04/03/07~~ 11

EXHIBIT B

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of		
ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	LicenseNo.
Entergy Nuclear Operations		DPR-26
Indian Point Energy Center Unit 2	)	Docket No. 50-247
License Renewal Application	)	

DECLARATION OF SHERWOOD MARTINELLI

My name is Sherwood Martinelli, and I reside with my wife and seven cats at 351 Dyckman Street in Peekskill, New York. The sinister twin domes of Indian Point are less than three miles from my residence, and can be viewed from the window of our attic sitting room when the leaves are off the trees. I am the founder of the Green Nuclear Butterfly, and Vice President of FUSE USA (Friends United for Sustainable Energy USA, INC. I have resided in Peekskill since the year 2000, my wife since 1996. As a landscaper, I spend a great deal of time interacting with a wide array of local citizens and business owners in Dutchess, Westchester and Rockland counties. My political involvements, including work as a volunteer for Congressman John Halls campaign have me out and visible in the community, my network of friends and business leaders a true pulse of our community. My recreational pursuits see me using the Hudson River for a place to walk, a place to boat (including canoe and kayak) as well as a place where I on occasion fish.

My hobbies include camping, hiking, biking, photography, and walking. A walk along the banks of the Hudson River down by the Peekskill Train Station is an enjoyable way to get some exercise, stretch my legs on a summers day, or chase away winters cabin fever after a Nor'easter has blown on out to sea and the snow melted away.

My concerns with the nuclear industry, and the NRC's inability to adequately police their licensees, date back to my years living in the foothills of South Eastern Ohio which lead up into the Blue Ridge Mountains. I founded the Save The Wills Creek Water Resources Committee when I uncovered six hundred thousand tons of Low Level Radioactive Waste that had been ILLEGALLY dumped into a wetlands that drained into my community's only drinking water supply. Imagine my shock and horror when I learned through investigation that the NRC had failed to police their licensee, Shieldalloy and their predecessors, for a period of over two decades.

I opposed the privatization of the two DOE Gaseous Diffusion Plants (in Portsmouth, Ohio and Paducah, Kentucky, and their transfer to a newly created company USEC which was given three hundred million in tax payer dollars as start up funds. More importantly, I opposed the transfer of jurisdictional control to the NRC, partially because of their deplorable and lack of enforcement.

When I first arrived in Peekskill, I was newly in love, and unaware that two aging leaking nuclear reactors were my neighbors, threatened my life, health and emotional well being. My first inkling came one day while out tending my flowers and the wailing sirens sounded the alarm. Entergy and the NRC are not really good at giving us warning when the tests take place, and half the time when we do manage too know about them the alarms fail in one sense or another. Then came the whisperings from neighbors of an evacuation plan that would not work which got me to begin taking an interest in the Indian Point site.

September 11th, 2001 was, or should have been, a wake up call to all of us. Sadly it did not seem to be, as the NRC seems far more intent in protecting their licensees than human health and the environment.

On September 11th, I was taking my wife to work at Bronx Community College when the report came in over the radio that a plane had crashed into one of the Twin Towers. We walked into the office to see other members of the staff huddled around a small black and white television set, just in time to see the second plane crash into the second tower. In unaminous shock, we all gasped knowing that our lives had forever changed and that America was under attack. I openly wept when the first tower crashed to the ground, and intuitively knew the second would soon follow suit.

The task of shutting down the college began, and duty interceded as my wife and others began implementing the necessary steps to close down and evacuate the campus. Out on the quad a stunned professor walked towards me with tears in his eyes. I came to learn that his son worked in one of the towers, and he could not reach him on the phone. Students were in a daze; some sat on benches crying alone, while others sat with friends, all with numb expressions.

Mid afternoon my wife and I headed home, an odd deathly pall hanging in the air. It was eerie driving up 87, and then on the Sprain. The only cars on the road were emergency vehicles from near and far, headed South into New York City. Four days later, still glued to the TV, exhaustion finally taking over I passed out curled up on the sofa in our living room. Now, six years to the day, I sit here in front of my laptop watching newsreels from the tragic day, and tears still sting my eyes as I write this declaration.

Entergy wants to relicense two aging nuclear relics for 20 more years of operation, telling us at every chance the Indian Point reactors are vital, safe and secure.

Strontium 90, Tritium and Cesium 137 leaks, sporadically working sirens, and sleeping guards speak volumes that drown out Entergy's pathetic lies. The NRC tells us the relicense of the facilities is an acceptable risk in the name of a greater societal good, and that the Emergency Evacuation Plan is adequate, but not up for discussion, and not considered within scope for the license renewal. If we bring up security, or the aftermath of a successful terrorist attack, we are scolded like children. We are told that the security at the plants is the best in the industry (which is not saying much), and that the odds of a terrorist attack are so small as to be almost non existent. This is why the DBT is off limits, and why the NRC contends that they are not required to consider the environmental costs associated with a terrorist attack in their Environmental Impact Statement.

Some look at my long hair, hear my anti-nuclear rhetoric and label me a lunatic. Others shrug their shoulders as if silently accepting their fate. I on the other hand have read, researched and found the lies that are the nuclear industry. Enforcement is not issuing exemption after exemption to your licensees that allow them to ignore rules and regulations created to keep us safe. Issuing Generic Letters alerting licensees of known serious equipment failure, yet requiring little or no mandatory action on the part of their licensees is not regulatory control. Inspections conducted by NRC staff are of no use and provide no incentive to abide by the rules, when every violation is written up as a green non-cited violation, even if the violation presents a serious risk to human health and the environment. A licensing process that lets licensees skirt the requirements of 10 CFR 54 by agreeing to a future list of commitments that they will file a letter requesting relief from is nothing more than a jury rigged system, a deceitful rape of communities being forced, with no real say, to continue hosting what are unsafe and dangerous facilities.

As a citizen who lives three miles from Indian Point, I should not be forced to live in fear, yet the continued operation of Indian Point leaves myself and 21 million other people living within 50 miles of the aging, embrittled reactors any other choice. Basic commonsense tells us that the evacuation plan is a farce. The Witt Report proves it beyond a shadow of a doubt. Rather than be truthful about this fact, Entergy and the NRC are now trying to reprogram public expectations, sell us on Sheltering in Place. Go to the Centers For Disease Control website, and you soon realize that Sheltering in Place is just another con game. The average citizen sheltered in a wood frame or brick home with a concrete basement is only afforded a 40 percent level of protection in the event of a nuclear attack. Eliminate the basement, and that level of protection drops to just ten percent. Entergy tells us it would be only days, and the NRC agrees. Problem is, the State Departments official website tells the real truth, and suggests we be prepared to shelter in place for as long as three weeks. The Department of Homeland Security harkens us to "be prepared," meaningless instructions evidently borrowed from the Boy Scouts.

If the Indian Point reactor and entire nuclear industry is so safe, why can't we as home owners get insurance to cover our losses should a nuclear incident or terrorist

attack occur? Could it have something to do with the overly hardened and brittle reactor cores that are now overly suspect to become victims of thermal shock, or the insurance industry's inside knowledge of what the devastation would be in the case of a successful terrorist attack on a nuclear reactor such as Indian Point? Is it fair to indemnify and hold the nuclear plant harmless in the case of and accident caused by industry and NRC negligence with the recent renewal of the Price Anderson Act?

. When Indian Point Two and Three were cited and built we were promised closed cooling systems, and even an 80 acre forested park with walking paths on the 235 acre site. More than 30 years later, we are still waiting on the closed water cooling system, and the park is just another broken promise. Acceptable risk, and what this risk encompasses should be a community decision, not an agency decision. If 50 percent plus one of us feels the risk of having Indian Point as a neighbor has become too great a risk, then the time to shut the reactors down is upon us. The NRC and Entergy will not accept the reality, but the time has come. Entergy's License Renewal Application should be denied to protect my health and safety, the health and safety of my family, and the health and safety of 21 million people who live within the 50 mile circle of death surrounding Indian Point in the event of a significant nuclear event.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 11th day of September, 2007, at Spring Valley, NY.

Sherwood Martinelli
Vice President FUSE USA
Distance from Indian Point: 8.5
Personal Contact information
351 Dyckman Street
Peekskill, New York 10566

914 734 1955

Witnessed and sworn to before
me ~~this 21~~rst day of
September, 2007

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 20 11

EXHIBIT C

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	)	**Docket**
License Renewal Application		**No. 50-247**

DECLARATION OF JULIE GOTTESMAN

My name is Julie Gottesman I live at 128 Highmount Avenue, Nyack, NY 10960. I am a member of Friends United for Sustainable Energy, USA, Inc (FUSE).

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in Nyack for approximately 6 years. I enjoyed walking and cycling along the river at Hook Mountain with my family. I am a mother of two children under 12 years old. I am currently a member of a rowing team, that practices in the Hudson River 2 to 3 times a week.

As a resident of Rockland County I am concerned that due to the limited water supply, the county is currently considering using the Hudson for drinking water.

I am aware that the plant is currently leaking dangerous radioactive contaminants from the plant into the ground around the plant, as well as the Hudson River.

In the event the river continues to be contaminated from releases from Indian Point, my enjoyment of the river for recreation and exercise will be directly affected. In addition, if Rockland County needs to start using the river for our public water supply my health and the health of my children may be adversely affected.

Indian Point needs to be shutdown, I understand the law requires the site to cleaned up to the condition it was in prior to the plant being built. It appears that the law is being broken. For example in the case of Unit 1, which was shut down over 30 years ago, its spent fuel pool is currently leaking Strontium 90, tritium and cesium into the river. The river is continuing to be polluted as a result of the inaction of the owners and regulators.

How can the NRC allow the operators continue operating a plant in this condition, let alone consider relicensing it for another 20 years?

If this was any other kind of business, such as a gas station, the government authorities would shut it down and make the owners remediate the leaks immediately.

Many of my neighbors and friends share this view, and are astonished at the apparent inability for the federal government to recognize this obvious lack of oversight to protect my health and my children's health.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 12th day of September 2007, at Nyack, NY.

Julie Gottesman

Sworn to before me this
12th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

EXHIBIT D

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	)	**Docket No. 50-247**
License Renewal Application	)	

DECLARATION OF GARY SHAW

My name is Gary Shaw, I live at 9 Van Cortlandt Place, Croton on Hudson, NY 10520. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE), Croton Close Indian Point (CrotonCIP), a member of the Steering Committee of the Indian Point Safe Energy Coalition (IPSEC.

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in the Hudson Valley for 15 years, and proximity to the Hudson was very important in our decision to move to Croton. The Hudson is a unique and vital resource to our community and the entire New York region. Today, Indian Point could not be cited in Buchanan, NY, due the enormous surrounding population and lack of a viable evacuation plan. The evacuation plan has been evaluated by a preeminent expert in emergency planning, James Lee Witt, and was judge inadequate and to a large degree, unfixable.

I am involved in Hudson River activities such as many Earth Day riverbank clean-ups during which I have often gotten abrasions and cuts while removing debris from the riverbanks. I have never before been concerned about my activities when pulling illegally dumped debris from the riverbank. Among the materials I have personally extracted are construction materials such as panels of house siding and aluminum window frame, car parts, tires and household appliances such as air conditioner and a refrigerator.

I am aware that the plant is allowed to discharge regulated amounts of radioactive elements into the river and that there are also currently unregulated leaks of radioactive contaminants from an undetermined number of sources into the ground around the plant, and that the contaminated water's pathway is generally towards the Hudson River. With the leakage continuing unabated and the potential for increased flow due to system degradation over time, my participation in river cleanups would have to be reevaluated.

Because this leakage is not yet directly linked to a known source of drinking water, the NRC has declared that the uncontrolled leaks are not a threat to public health or safety. As a user, but not a drinker of the river, I am concerned.

The NRC is considering granting the plant a license renewal that will result in twenty more years of high level nuclear wastes that will also go into spent fuel storage that is leaking now and will continue to leak long after the plant has finally been decommissioned. I am concerned that my health may be compromised because Indian Point currently is and will apparently be allowed to leak radioactivity indefinitely. In fact, since the spent fuel pool at Indian Point 1 is believed to be among the sources of leakage, and Indian Point 1 has been inactive for decades, it appears that the plant will leak into perpetuity. That would appear to be a preview of the future of Indian Point 2. Allowing 20 more years of additional wastes to be generated and stored in leaking pools seems to me to be a direct threat to citizens' health and safety.

If this were any other kind of business, such as a gas station, wouldn't the EPA or other regulatory agency shut it down and make the owners remediate the leaks immediately?

It is unacceptable for the NRC to allow Indian Point to continue to contaminate the groundwater and the Hudson River. I am certain that many of my neighbors and friends share this view, as evidenced by the wide-spread willingness to sign petitions in opposition to Indian Point at each year's Croton Village Summerfest, and actions by the village Board of Trustees, including passing resolutions supporting the congressional call for an Independent Safety Assessment, and previously calling for plant closure and opposition to relicensing.

The public's health and safety should not be compromised for the financial benefit of a privately owned corporate polluter, whose parent company has allowed the bankruptcy of another of its nuclear plants in order to avoid financial liabilities in the aftermath of Hurricane Katrina. In addition, many articles have suggested that energy efficiency and conservation programs, and the upgrading of the deteriorating transmission lines would mitigate the perceived need for Indian Point's electrical output.

I declare that the statements made in this declaration are true and correct to the best of my knowledge.

Executed this 15th day of September, 2007, at Croton on Hudson, NY.

Gary Shaw

Sworn to before me this
15th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 20 11

EXHIBIT E

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	**)**	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	**)**	**Docket No. 50-247**
License Renewal Application	**)**	

DECLARATION OF ANDREW Y. STEWART, PhD

My name is Andrew Y. Stewart, I live at 19 Mill Street, Nyack, NY 10960. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE) and the Executive Director of Keep Rockland Beautiful, Inc.

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in the Hudson Valley for 17 years, and am very connected to the Hudson River. About 12 years ago I built my own kayak and use it in the Hudson. Also I have built my own small sail boat.

I teach environmental science at Rockland Community College. For the past 6 years I have organized volunteer clean-up of the banks of the Hudson. For the past 3 years I have helped Hudson River Basin Watch put together educational workshops for high school students on the Haverstraw water front, regarding land use and water quality.

Rockland County is currently considering using the river for tap water, due the limited water resources in the county.

The Hudson River is a unique and vital resource to our community and the entire New York region. Today, Indian Point could not be cited where it is currently located, due the enormous surrounding population and lack of a workable evacuation plan.

Indian Point is currently leaking radioactive waste into the groundwater and River, yet the NRC is considering to permit it to continue operating and leaking for another 20 years.

It is unacceptable for the NRC to allow Indian Point to continue to contaminate the groundwater and Hudson.

If the NRC permits Entergy to continue operation of this aging plant that is polluting the River, it will directly affect my lifestyle by preventing me from enjoying the river for exercise and will stop me from being able to bring students and community members to its banks. In addition it may directly affect the health and safety of my family.

The public's health and safety cannot be compromised, for the sole benefit of a privately owned corporation.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 10 day of September, 2007, at Nyack, NY.

Andrew Y. Stewart

Sworn to before me this
10th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

EXHIBIT F

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, LLC	)	**LicenseNo. DPR-26**
And Entergy Nuclear Operations, Inc.	)	**Docket**
Indian Point Energy Center Unit 2		**No. 50-247**
License Renewal Application	)	

DECLARATION OF TIMOTHY ENGLERT

My name is Timothy Englert. I live at 260 Mirth Drive, Valley Cottage, NY 10989, with my wife and two young children, within in 10 miles of Indian Point. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE) and work for the Palisades Interstate Parks Commission.

I have lived in the Hudson Valley for 7 years, and in both my private and professional life am very involved with the Hudson River. I am an avid kayaker, canoeist and crew rower, and am on the water a minimum of one day a week from the spring through the fall. I use the Hudson river from Bear Mountain down through Piermont.

The Hudson River and Hudson Valley is incredibly beautiful and is vital resource to our community and the entire New York region. As Development Specialist for the Palisades Interstate Park Commission, I am tasked with creating recreational, educational, and philanthropic opportunities with our parks, many of which lie directly on the Hudson River

I try to avoid going near Indian Point when I am kayaking because of the thermal pollution and growing concern about the radioactive leaks into the river.

I understand that the owners of Indian Point, Entergy, have applied for a new license for 20 more years. I cannot see how the NRC can possibly approve this, when every other day there is some problem reported in our local newspapers about Indian Point, including radioactive leaks, fires, and siren problems.

The population in the Hudson Valley is extremely dense and the road infrastructure is very limited. An accident on the New York State Thruway or the Tappan Zee Bridge causes the traffic to stop for hours. If something happened at Indian Point, evacuation from this area would be nearly impossible.

I understand that Rockland County is currently considering using the river for tap water, due the limited water resources in the county. I am concerned that the ongoing leaks into the Hudson River could have health affects on me and my family.

It is unacceptable for the NRC to allow Indian Point to continue to contaminate the groundwater and Hudson river now, let alone for another 20 years.

If the NRC permits Entergy to continue operation of this aging plant that is polluting the River, it will directly affect my lifestyle and that of my children, especially as they embrace the waters of the Hudson.

The public's health and safety cannot further be compromised, for the benefit of a privately owned corporation.

Friends United for Sustainable Energy USA, Inc., (FUSE) represents me in the above cited Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 19th day of September, 2007, at Valley Cottage, NY.

Timothy Englert

Timothy Englert

Sworn to before me this
19th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

EXHIBIT G

NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2 **License Renewal Application**	)	**Docket No. 50-247**

DECLARATION OF JEANNE SHAW

My name is Jeanne Shaw, I live at 9 Van Cortlandt Place, Croton on Hudson, NY 10520. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE).

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in the Hudson Valley for 15 years. I am an artist who uses driftwood from the Hudson River for many of my pieces. I have spent and continue to spend much time walking along the banks of the Hudson collecting materials for my work.

I am aware that the Indian Point Nuclear Power Plant is currently leaking dangerous radioactive contaminants into the ground around the plant, and that the general flow of the contamination is towards and into the Hudson River. While publicized testing and off-site readings indicate that my beachcombing is currently uncompromised and my art materials are contaminant free, I am concerned that continuing leakage, especially if the aging process leads to faster or larger leaks, will affect to continue my work and interfere with my access to river materials.

I believe the law requires industrial sites to be cleaned up and restored to the condition they were in prior to the plant being built. It appears that the law is being ignored, since Indian Point Unit 1, which was shut down over 30 years ago, is currently leaking Strontium 90, tritium and cesium into the surrounding environment and subsequently, into the river. The river is continuing to be polluted as a result of the inaction of the owners and regulators.

How can the NRC allow the operators to continue operating a plant in this condition, let alone consider relicensing it for another 20 years? If this were any other type of industrial or business site, such as a gas station with leaking tanks or a dry cleaner allowing toxic chemicals to enter the environment, wouldn't either state or federal regulatory authorities shut it down and make the owners remediate the leaks immediately? It is unacceptable for the NRC to allow Indian Point to continue to contaminate the groundwater and the ultimately the Hudson River.

The public's health and safety should not be compromised for the financial benefit of a privately owned corporation. I believe that many of my neighbors and friends share the view that tacit acceptance of radioactive leaks by the federal government's regulators represents a very limited perspective of what constitutes a threat to public health and safety.

I declare that all statements in this declaration are true and correct to the best of my knowledge.

Executed this 15th day of September, 2007, at Croton on Hudson, NY.

Jeanne D Shaw

Jeanne Shaw

Sworn to before me this
15th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

EXHIBIT H

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2 **License Renewal Application**	)	**Docket No. 50-247**

DECLARATION OF ROBERT A. JONES

My name is Robert A. Jones, I live at 124 Trails End, New City 10956, with my wife and my three young children. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE).

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in Rockland for 38 years. Until a few years ago I used the river for swimming and water skiing, off my boat. I stopped due to the condition of the water and all the leaks you hear about from Indian Point.

When I was swimming and waterskiing, my friends and I would park our boats just north of the Haverstraw Bay, and we noticed the dramatic different in water temperature. It was always much warmer there. When we learned that is was warmed because Indian Point was dumping heated water into the river we immediately using it, and it turned me off from swimming any where in the river. Indian Point has changed my quality of life.

Now that I know that strontium and tritium in to the river I am even more concerned.

I heard that they are considering using the Hudson River for Rockland County tap water, I think its crazy. Certainly if I won't swim it I wont' drink it or bathe in it. Or permit my young children to do so. This will certainly affect my quality of life.

I love the Hudson River because it is beautiful area, it is close to home, convenient, unfortunately because of Indian Point there are many things I used to do on the water, that I now cannot do.

drink it or bathe in it. Or permit my young children to do so. This will certainly affect my quality of life.

I love the Hudson River because it is beautiful area, it is close to home, convenient, unfortunately because of Indian Point there are many things I used to do on the water, that I now cannot do.

The Hudson River is a unique and vital resource to our community and the entire New York region. Today, Indian Point could not be cited where it is currently located, due the enormous surrounding population and lack of a workable evacuation plan.

I work for a company that owns a gas station, where a spill was reported, the DEC and Health Department immediately shut down the station, until it was totally dug up and remediated, even though it turned out not to be the gas stations fault. I cannot understand how our government allows Indian Point to remain open and be considered for relicensing for another 20 years, with all the leaks and problems that keeps arising.

Indian Point is currently leaking radioactive waste into the groundwater and River, yet the NRC is considering to permit it to continue operating and leaking for another 20 years, to me this totally insane.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 12 day of September, 2007, at Spring Valley, NY.

Robert A. Jones

Sworn to before me this
12th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

EXHIBIT I

NOT LOCATED

LAW OFFICES OF

LEBOEUF, LAMB, LEIBY & MACRAE

1821 JEFFERSON PLACE, N.W.

WASHINGTON, D.C. 20036

October 15, 1968

Dr. Peter A. Morris
Director
Division of Reactor Licensing
U.S. Atomic Energy Commission
Washington, D. C. 20545

Re: AEC Docket No. 50-247

Dear Doctor Morris:

Transmitted herewith are three (3) originals and nineteen (19) conformed copies of Amendment No. 9 to the Application for Licenses in the above-captioned proceeding, together with seventy-three (73) copies of the technical material referred to therein.

Copies of these documents will be served today upon Mr. William J. Burke, Mayor, Village of Buchanan, New York, and a Certificate of Service will be filed with the Commission later today.

Sincerely yours,

LeBoeuf, Lamb, Leiby & MacRae
Attorneys for Consolidated
Edison Company of New York, Inc.

Enclosures

RECEIVED
OCT 15 1968

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	Docket No. 50-247
Consolidated Edison Company of New York, Inc.	)	
	)	

Amendment No. 9

to

Application for Licenses

Consolidated Edison Company of New York, Inc., Applicant in the above-captioned proceeding, hereby files Amendment No. 9 to its Application for Licenses for the purpose of transmitting its Final Facility Description and Safety Analysis Report, consisting of four volumes.

WHEREFORE, Applicant prays as in its original Application for Licenses.

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

By *[signature]*
W. Donham Crawford
Administrative Vice President

Dated: *October 11, 1968*

Subscribed and sworn to before me this *11th* day of *October*, 1968.

Frances E. Flynn
Notary Public

My Commission Expires *March 30, 1969*.

FRANCES E. FLYNN
Notary Public, State of New York
[illegible]
Qualified in Kings County
Cert. Filed in New York County
Commission Expires March 30, 1969

1.3 GENERAL DESIGN CRITERIA

The general design criteria define or describe safety objectives and approaches incorporated in the design of this plant. These general design criteria, tabulated explicity in the pertinent systems section in this report, comprise the proposed Atomic Industrial Forum versions of the criteria issued for comment by the AEC on July 10, 1967. The remainder of this section, 1.3, presents a brief description of related plant features which are provided to meet the design objectives reflected in the criteria. The description is developed more fully in those succeeding sections of the report indicated by the references.

The parenthetical numbers following the section headings indicate the numbers of its related proposed General Design Criterion (GDC).

1.3.1 OVERALL PLANT REQUIREMENTS (GDC 1-GDC 5)

All systems and components of the facility are classified according to their importance. Those items vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity are designated Class I. Those items important to reactor operation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity are designated Class II. Those items not related to reactor operation or safety are designated Class III.

Class I systems and components are essential to the protection of the health and safety of the public. Consequently, they are designed, fabricated, inspected and erected and the materials selected to the applicable provisions of recognized codes, good nuclear practice and to quality standards that reflect their importance.

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EXHIBIT J

5

Mrs. Becker

DOCKET NUMBER PROPOSED RULE PR-50
General Design Criteria

ATOMIC INDUSTRIAL FORUM INC.

850 THIRD AVENUE · NEW YORK, N.Y. 10022 · PLAZA 4-1075

October 2, 1967

DOCKETED USAEC OCT 4 1967 Office of the Secretary Public Proceedings Branch

Secretary
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Sir:

Pursuant to notice which appeared in the Federal Register of July 11, 1967, the Forum Committee on Reactor Safety is pleased to forward the enclosed comments on AEC's proposed "General Design Criteria for Nuclear Power Plant Construction Permits".

These comments, which in a number of instances take the form of a redraft of the proposed criteria, are based on information developed during an August 9 meeting of the Committee. They have been further refined by a Committee task force comprised of the following members: Wallace Behnke of Commonwealth Edison Company; Arthur C. Gehr of Isham, Lincoln & Beale; R. J. McWhorter of General Electric Company; J. E. Tribble of Yankee Atomic Electric Company; Robert A. Wiesemann of Westinghouse Electric Corporation; and Edwin A. Wiggin of the Forum staff.

The comments have subsequently been circulated to those additional members of the Committee who participated in the August 9 meeting. It may, therefore, be concluded that the enclosed comments generally represent the views of the following additional Committee members:

R. H. Bielecki, Pennsylvania Power & Light Company
Warren S. Brown, Dilworth, Secord, Meagher & Associates, Ltd.
Harvey F. Brush, Bechtel Corporation
Robert W. Davies, Baltimore Gas and Electric Company
William S. Farmer, Allis-Chalmers Manufacturing Company
George C. Freeman, Jr., Hunton, Williams, Gay, Powell & Gibson
Robert E. Kettner, Consumers Power Company
R. W. Kupp, S. M. Stoller Associates
C. A. Larson, Consolidated Edison Company of New York, Inc.
Zelvin Levine, Hittman Associates, Inc.
James V. Neely, Jersey Central Power and Light Company
H. C. Ott, Ebasco Services, Inc.
Joseph W. Ray, Battelle Memorial Institute
Glenn A. Reed, Wisconsin Electric Power Company
Marlin Remley, Atomics International, Inc.
Royce J. Rickert, Combustion Engineering, Inc.

c/3

W. N. Thomas, Virginia Electric and Power Company
Robert E. Wascher, The Babcock & Wilcox Company
Samuel Zwickler, Burns & Roe, Inc.

Although these comments have been throughly reviewed by those individuals listed above, it should be understood that they do not necessarily represent a unanimity of opinion on all the criteria. Members of the Committee who participated in the August 9 discussion, particularly those who find themselves at variance with the views expressed herein, have been urged to make their views known directly to the AEC in behalf of their own respective companies and organizations.

Perhaps a further note of explanation on the enclosed comments is in order.

In the Committee's opinion, the proposed criteria are appreciably better organized than those initially suggested in November 1965. We have also noted with appreciation that some of the Committee's suggestions on the earlier criteria have been accommodated in the criteria now proposed.

The Committee believes that the principal objectives of the criteria should be to assist in the design of nuclear power plants, the preparation of applications for construction permits and operating licenses therefor and regulatory review of these applications to determine if such plants can be constructed and operated without undue risk to the health and safety of the public. The Committee further believes that these objectives should be explicitly stated and that they can be most effectively attained by writing the criteria to the extent possible as performance specifications.

We recommend that the following paragraph be added to the introduction - possibly following the last paragraph of the introduction as it appeared in the Federal Register notice:

> "Each of the requirements stated and implied in the criteria is premised on assuring that the nuclear power plant will be designed, constructed and operated in such a manner as not to cause undue risk to the health and safety of the public from radiation or the release of radioactive materials. To facilitate compliance with the requirements contained in the criteria, the criteria are presented to the extent possible, as performance specifications."

The Committee further believes that the introduction to the criteria should make more explicit reference to their intended direct applicability to water reactors in contrast to their only indirect applicability to reactors of other types, including fast breeders.

Some members of the Committee have noted the desirability and advantages of publishing these criteria as a guide rather than as an appendix to 10 CFR 50. They point out that, as a guide, their interpretation, application and refinement could be more easily adapted to a rapidly

ATOMIC INDUSTRIAL FORUM INC.

If questions arise in reviewing these comments, the members of the task force would be pleased to meet with representatives of the AEC regulatory staff.

Sincerely,

Edwin A. Wiggin

Edwin A. Wiggin
Committee Secretary

EAW:epb
Enclosure

PROPOSED RULE R-50
General Design Criteria

Comments of Forum Committee on Reactor Safety on AEC's Proposed Construction Permit Criteria

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.

> In the first sentence we have modified "accidents" with "nuclear" and substituted the phrase "cause undue risk to the health and safety of the public" to more precisely reflect what we believe was the AEC's intent. In the last sentence of the original draft, we have dropped the word "sufficiency" since we do not believe that it should be the responsibility of the applicant to document this unless the sufficiency of some specific item is in question. If for any reason the AEC questions the adequacy or sufficiency of a code or standard, it should take this matter up with the appropriate code drafting committee. Note that we have added a sentence requiring a showing of adequacy where there is no applicable code. The balance of the suggested changes are editorial in nature.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

> The changes in the first sentence are in line with those suggested for Criterion 1. We have deleted the word "additional" on the premise that it is not reasonable to ask the applicant to consider the simultaneous or cumulative forces of more than one extraordinary natural phenomenon.

CRITERION 3 - FIRE PROTECTION (Category A)

A reactor facility shall be designed such that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

> These changes are consistent with the objective of assuring that there will be no undue risk to the health and safety of the public.

CRITERION 4 - SHARING OF SYSTEMS (Category A)

Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public.

As originally drafted, this criterion made unacceptable any impairment of safety, whether the impairment was significant or insignificant. This is unreasonable. Some impairment will undoubtedly result from almost any sharing but the impairment may not be significant enough to preclude the sharing. The test should be whether the sharing will result in undue risk to the health and safety of the public.

CRITERION 5 - RECORDS REQUIREMENTS (Category A)

The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public.

Some of the records that should be maintained may or may not be under the physical control of the licensee or operator. He can, however, assure that they are maintained, by contractual arrangements, if necessary. Those records which are important are those which could have some bearing on the health and safety of the public.

CRITERION 6 - REACTOR CORE DESIGN (Categories A & B)

The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

We assume that "acceptable fuel damage limits" will be based on "undue risk to the health and safety of the public", not on economic grounds. The latter consideration is a matter for the licensee to decide. Further, these limits will depend on the circumstances leading to the damage. The example "transient situations" have been deleted since they may not be applicable in certain cases and they might also tend to prejudice design innovations.

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

See comment on Criterion 6 with respect to "acceptable fuel damage limits".

CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

We recommend deletion of this criterion since it is not applicable to certain reactor types. It is possible for the overall power coefficient resulting from a sum of components with different time constants to be positive without causing any serious safety problem. For example, in a sodium graphite reactor the coefficient has a prompt negative component together with a positive component with a long time constant. This results in an overall positive coefficient, but the negative part of the coefficient is large enough and fast enough to assure satisfactory control and safety. Safety problems relating to reactivity considerations are adequately covered in Criteria 6 and 7.

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

It is important to characterize the leakage as "uncontrolled". Our only other suggested change is insertion of the word, "fabricated".

CRITERION 10 - REACTOR CONTAINMENT (Category A)

Reactor containment shall be provided. The containment structure shall be designed (a) to sustain without undue risk to the health and safety of the public the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public.

To avoid any ambiguity, "containment" should be characterized as "reactor containment". The statutory requirement of the licensee and the AEC is "to avoid undue risk to the health and safety of the public", not "to protect the public". It would be helpful to cross reference this criterion to Criterion 37 to indicate what the AEC means by "engineered safety features". Consistent with our comments on Criterion 37, we have substituted "pipe" for "boundary" on the premise that an applicant should not be required to consider a design basis accident more conservative than the instantaneous double-ended, circumferential rupture of a large coolant pipe.

CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel.

As originally drafted, this criterion could be interpreted as requiring a second control room. Not only would such a requirement be inconsistent with current practice, we believe that the complexities introduced could adversely affect overall plant safety. We believe it possible to design and equip a control room to assure continuous occupancy under all circumstances, including fire. We have deleted reference to 10 CFR 20 since the radiation exposure limits set forth therein apply to normal operating conditions, not accident conditions. Compliance with the radiation exposure limits of 10 CFR 20 under accident or post-accident circumstances is neither necessary nor reasonable. We have deleted the last sentence of the original draft since it is unnecessary and contradictory with the requirement of continuous occupancy of the control room.

CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables.

We have modified this criterion to more accurately and precisely reflect its intent.

CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core.

We have dropped the two examples since they are measures of reactivity rather than the fission process.

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

We have deleted the phrase "act automatically" since manual action will prove adequate, indeed desirable, in some instances.

CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

No change suggested.

CRITERION 16 - MONITORING REACTOR COOLANT LEAKAGE (Category B)

Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.

We have assumed the intent of this criterion is to assure that leakage from the primary system will be detected, not that the entire reactor coolant pressure boundary will be monitored. The latter requirement would be inconsistent with current practice and unnecessary. Also, consistent with Criterion 9, we believe that the leakage should be characterized as significant and uncontrolled.

CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (Category B)

Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive.

We believe that the modified language as indicated above more accurately and precisely reflects the intent of the criterion.

CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (Category B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels.

We believe that the modified language as indicated above more accurately and precisely reflects the intent of the criterion.

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public.

The suggested change is in line with our comment on Criterion 1.

CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (Category B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served.

The significant change we have made here is to delete the last sentence of the original draft. It would appear preferable to provide duplicates of the best system or component rather than going to an inferior system or component based on a different principle.

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

We recommend deletion of this criterion since it is more of a definition than a criterion and since the implied requirement is adequately covered by Criterion 23.

CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

This criterion should be deleted inasmuch as its requirements, to the extent they should be included in general criteria, ... be adequately covered by Criterion 20

CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function or shall be tolerable on some other basis.

> The suggested change here includes adding to the criterion the phrase, "or shall be tolerable on some other basis".

CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (Category B)

> We recommend deletion of this criterion since it would appear preferable to focus all requirements for emergency power in Criterion 39. Note that "protection systems" has been incorporated in Criterion 39 to accommodate this deletion.

CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred.

> The reason for the changes here is that the licensee should be given some latitude in determining when and how such tests should be carried out. Further, he should be required only to test the active components of a protection system in contrast, for example, to a rupture diaphragm which could only be tested at the expense of destroying it. Also, certain tests might permit the licensee to determine if failure or loss of redundancy has occurred, but they might not permit him to demonstrate it.

CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

> No change suggested.

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

Two independent reactivity control systems, preferably of different principles, shall be provided.

> The phrase, "At least" which prefaced the original criterion suggests a possible escalation of requirements which we do not believe was intended.

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition.

> Deletion of the preface phrase, "At least two of" is based on the comment made on Criterion 27. We have deleted the examples at the end of the original criterion since they could be interpreted to indicate a requirement for two fast reactivity shutdown mechanisms. This requirement is unnecessary when there is sufficient redundancy in one of the reactivity control systems to assure shutdown.

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.

> Deletion of the preface phrase, "At least", is consistent with the comments on Criteria 27 & 28. The other editorial changes are for purposes of clarification.

CRITERION 30 - REACTIVITY HOLDOWN CAPABILITY (Category B)

The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

> Deletion of the preface phrase, "At least one of", is consistent with the comments on Criteria 27, 28 & 29. Further, the public health and safety will not be compromised by a return to low power.

CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

> We believe the criterion should preserve its original objective and at the same time acknowledge that one of the functions of the reactor protection system is to protect against certain control system malfunctions.

<u>CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)</u>

Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.

> We believe substitution of "reasonable" for "considerable" and the substitution of "lose capability of cooling the core" for "impair the effectiveness of emergency core cooling" more precisely reflects the intent of the criterion. The re-wording also correctly implies that emergency core cooling will generally be required only if the reactor coolant pressure boundary is breached.

<u>CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)</u>

The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

> We have deleted the phrase, "and with only limited allowance for energy absorption through plastic deformation", on the premise that it is not helpful.

CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failures. Consideration shall be given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

The detailed requirements contained in the original version are not appropriate for general criteria.

CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

With the re-writing of Criterion 34 as indicated above, this criterion can and should be deleted.

CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided.

It should not be necessary to inspect or maintain surveillance over all portions of the coolant pressure boundary; hence, we have inserted the phrase, "of critical areas". We believe that both the applicant and the AEC are in a better position to take advantage of developing technology and code refinement if these general design criteria refer to "current applicable codes" rather than to specifically designated codes.

CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

> Deletion of the phrase, "As a minimum", and substitution of "piping" for "pressure boundary" are both intended to eliminate the implication that the applicant should be required to consider a design accident basis more conservative than the instantaneous, double-ended, circumferential rupture of the largest pipe in the primary system. On this premise, retention of the original language introduces a vagueness which tends to defeat the objective of the criterion.

CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (Category A)

All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public.

> Avoiding undue risk to the health and safety of the public is the purpose of all engineered safety features and the "functional reliability and ready testability" of such features is directly related to their attainment of this objective. To tie this criterion to the problem of siting appears extraneous and not helpful; hence, we have deleted the second sentence.

CRITERION 39 - EMERGENCY POWER (Category A)

An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single active component.

> As originally drafted, this criterion could be interpreted as requiring two off-site and two on-site power sources. Since neither the AEC nor the licensee may have any control over

the off-site power supply and since an emergency on-site power supply adequate to meet the power needs of the engineered safety features is required, any reference to off-site power is irrelevant. We have, therefore, re-written this criterion to eliminate such reference to off-site power. We have also changed the title of the criterion to accommodate the addition of "protection systems", which reference was added because of the deletion of Criterion 24.

CRITERION 40 - MISSILE PROTECTION (Category A)

Adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures.

The suggested changes in this criterion are for purposes of clarification.

CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features such as the emergency core cooling system and the containment heat removal system shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

We believe the measure of "sufficient performance capability" of an engineered safety feature should be that no undue risk to the public health and safety will result from the failure of any single active component of that feature. The modified language, in our opinion, more accurately and precisely reflects the intent of the criterion.

CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public.

Although it would appear extremely difficult, if not impossible, to design engineered safety features in such a way that a loss-of-coolant accident will cause no impairment of the capability of any component or system, it is possible to design them to meet the requirements of this criterion as stated above.

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Protection against any action of the engineered safety features which would accentuate significantly the adverse after-effects of a loss of normal cooling shall be provided.

> The intent here was simply to state the criterion in a more positive way.

CRITERION 44 - EMERGENCY CORE COOLING SYSTEM CAPABILITY (Category A)

An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

> In our opinion, one emergency core cooling system which incorporates a sufficient redundancy of active components and covers the full range of postulated breaks should be adequate. Our modification of this criterion reflects this consensus. For this reason, we have omitted the last sentence of the original criterion.

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM (Category A)

Design provisions shall where practical be made to facilitate physical inspection of all critical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles.

> Since inspection of water injection nozzles is not always possible on a reasonably complete and non-destructive basis and since the failure of a safety injection nozzle is assumed in most accident analyses, we have inserted the phrase, "where practical".

CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEM COMPONENTS (Category A)

> No comment other than the criterion should be presented in the context of a single emergency core cooling system, consistent with the comments offered on Criterion 44.

CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEM (Category A)

A capability shall be provided to test periodically the operability of the emergency core cooling system up to a location as close to the core as is practical.

> Testing the "operability" in contrast to the "delivery capability" of the emergency core cooling system "up to" rather than "at" a location close to the core more accurately reflects the art of the possible and should provide for as adequate a test of reliability.

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEM (Category A)

A capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the emergency core cooling system into action, including the transfer to alternate power sources.

> The only change here, and a significant one we believe, is insertion of the word, "initially". Although we concur that a capability to test the operational sequence of the emergency core cooling system should be provided, the test as a practical matter would not be carried out frequently and possibly not more than once - prior to startup.

CRITERION 49 - REACTOR CONTAINMENT DESIGN BASIS (Category A)

The reactor containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system, will not result in undue risk to the health and safety of the public.

> The objective of this criterion, in our opinion, should be that under the circumstances of an accident the integrity of the containment should be such as to prevent

Undue risk to the health and safety of the public. Since the maintenance of containment integrity is based on effective functioning of the emergency core cooling system, it appears unreasonable in this criterion to assume the complete failure of the emergency core cooling system; hence we have assumed a failure of a single active component. Consistent with this assumption, we believe that the pressure and temperature to be withstood should be characteristic of those anticipated from the largest credible energy release associated with a loss-of-coolant accident, including the calculated energy from metal-water and other chemical reactions. Acceptance of the "failure of a single active component" concept is consistent with Criterion 41.

CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

The selection and use of containment materials shall be in accordance with applicable engineering codes.

It appears to us that the specific requirements of this criterion as originally drafted are not in keeping with the intent of general design criteria.

CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

If part of the reactor coolant pressure boundary is outside the containment, features shall be provided to avoid undue risk to the health and safety of the public in case of an accidental rupture in that part.

It is our understanding that it is the responsibility of the licensee to "avoid undue risk to" rather than "to protect" the health and safety of the public. We have deleted the second sentence of the criterion as originally drafted on the premise that it is only incidental to the requirement set forth in the first sentence.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure this system shall perform its required function, assuming failure of any single active component.

Deletion of the phrase "at least" is consistent with our comment on Criterion 27. The other changes are consistent with our comments on Criterion 41.

CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

No change suggested.

CRITERION 54 - INITIAL LEAKAGE RATE TESTING OF CONTAINMENT (Category A)

Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance.

We have inserted "initial" in the title to differentiate Criterion 54 from Criterion 55. Further, we believe it more realistic to leak test at peak pressures associated with postulated accidents than at design pressure. Correlation of leakage rate tests at postulated accident pressures with those conducted at design pressure prior to installation of containment penetrations will permit extrapolation of observed leakage rates to design pressure conditions.

CRITERION 55 - PERIODIC CONTAINMENT LEAKAGE RATE TESTING (Category A)

The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime.

Our suggested changes here are consistent with our comments on Criterion 54. Further, a requirement calling for periodic leak testing at design pressure would impose an unnecessary and impractical design requirement on the plant.

CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

Provisions shall be made to the extent practical for periodically testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident.

We have inserted the word, "periodically" to avoid an interpretation that we do not believe was intended, namely a requirement for "continuous" testing. The other suggested change is consistent with our comments on Criteria 54 & 55.

CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

> Our only suggested change here is insertion of "to the extent practical". We believe this is consistent with the intent of the criterion as originally drafted, but we also believe that the qualification should be explicit rather than implicit. This comment also applies to Criteria 58, 59, 60, 62, 63, 64 and 65.

CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

> See comment on Criterion 57.

CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS (Category A)

> See comment on Criterion 57.

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (Category A)

A capability shall be provided to the extent practical to test periodically the operability of the containment spray system at a position as close to the spray nozzles as is practical.

> Insertion of the phrase, "to the extent practical" is consistent with our comment on Criterion 57. The basis for substitution of "operability" for "delivery capability" is the same as that used in our comments on Criterion 47.

CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure-reducing systems into action, including

[illegible]

CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (Category A)

See comment on Criterion 57.

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (Category A)

See comment on Criterion 57.

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (Category A)

See comment on Criterion 57.

CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

See comment on Criterion 61.

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

No change suggested.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which would result in undue risk to the health and safety of the public.

We have substituted "which would result in undue risk to the health and safety of the public" for "to plant operating areas or the public environs" since we believe the first phrase more accurately describes the responsibility of the licensee.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category B)

Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities.

The suggested change permits the criterion to accommodate radiation limits as may be specified which may differ from those set forth in 10 CFR 20.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity.

> We have avoided the use of the word, "containment" because of its possible ambiguous connotation. The licensee may rely on some means other than containment to meet the requirements of the criterion. The other suggested changes are consistent with our comments on Criterion 67.

CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence.

> We have deleted the qualification on condition (b) namely, "except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents". This qualification is not helpful and could be subject to misinterpretation by the uninformed public.

basis that the core reactor internals remain functional and that adequate shut down margin can be achieved by control rod insertion, we conclude that the stress and deflection limits for the combined blowdown and design basis earthquake loadings provide an adequate margin of safety.

The primary system side of the steam generators, the pressurizer, and the main coolant pump casings, have been designed to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition - Summer 1969 Addenda, as Class A vessels. For other Class I pumps, valves, and heat exchangers the inspection program required independent review of (1) the physical and chemical test data for pressure boundary materials, (2) radiographs of valve bodies, valve bonnets and pump casings, and (3) dye-penetrant examinations of heat exchanger tubes and welds. These requirements resulted in fabrication and inspection programs that contain the essential elements of the recently proposed ASME Codes for Nuclear Pumps and Valves. We find the design codes and inspection requirements acceptable.

We have reviewed the information submitted by the applicant with respect to operating limitations on heatup and cooldown of the primary system imposed by the fracture toughness properties of the materials of the Indian Point Unit 2 reactor vessel. Our evaluation was based on a proposed redraft of section NB-2300 Special Materials Testing (Section III ASME Boiler and Pressure

Vessel Code) dated July 28, 1970, which reflects the material testing requirements in a form consistent with the AEC Fracture Toughness Criteria. As a consequence of our evaluation the applicant has agreed to the heatup and cooldown limitation as presented in Section 3.1-B of the Technical Specifications which represents a modification of his initial submittal. On the basis that these limits reflect a very conservative method of defining pressure vessel fracture toughness, we conclude that they are acceptable.

5.3 Coolant Piping

The reactor coolant piping has been designed in accordance with the requirements of the American National Standards Institute (ANSI) B31.1 Code for Power Piping, 1955 Edition, including the requirements of Nuclear Code Cases N-7 and N-10. All welding procedures and operators were qualified to the requirements of Section IX of the ASME Boiler and Pressure Vessel Code. Additional inspection requirements for the reactor coolant piping during fabrication included ultrasonic and dye-penetrant inspection of all pipe welds. Non-destructive examination of valves included radiographic examination of the valve castings and ultrasonic inspection of all forged components. Dye-penetrant surface examination was also performed. With this program, the inspection of the Indian Point Unit 2 reactor coolant piping substantially meets the requirements of Class 1 systems under the ANSI B31.7 Code for Nuclear Power Piping adopted in 1969. On this basis we have concluded that the design and inspection program for this system is acceptable.

The original seismic design analysis for the Indian Point Unit 2 reactor coolant system utilized only static methods of analysis. Recently, at our request, the applicant completed a rigorous dynamic analysis of this system utilizing both modal-response spectra and model time-history methods of analyses. As with the reactor internals, the combined loading of a concurrent loss-of-coolant accident blowdown and design basis earthquake was not considered in the design of the Indian Point Unit 2 reactor coolant system. However, the applicant recently completed an analysis of the response of the reactor coolant system to be installed in Indian Point Unit 3 for these combined loads. Since the Indian Point Unit 3 and the Indian Point Unit 2 reactor coolant systems are identical, the applicant has used the results of the analysis for Indian Point Unit 3 in conjunction with the material properties for the Indian Point Unit 2 piping, as determined from tests, to determine that the combined seismic and accident loads can be tolerated by the Indian Point Unit 2 reactor coolant system within acceptable stress limits.

Based on our review of the design limits and analytical procedures employed, we find that the design of the Indian Point Unit 2 reactor coolant system is acceptable.

5.4 Other Class I* (Seismic) Piping

At our request the applicant performed additional seismic analysis on other Class I piping. The adequacy of the seismic design of the feedwater lines, pressurizer surge line, and a typical steam line has been confirmed by a dynamic analysis utilizing the modal-response-spectra method. The adequacy of the seismic design of other Class I (Seismic) piping in the plant was determined by performing a dynamic analysis on selected "worst case" systems. Several systems that are the most vulnerable to dynamic excitation because of system flexibility or location in the supporting structure were analyzed and the resulting stresses compared with the stresses determined by the original static analyses. The applicant has concluded that the conservatism of the original static analysis provided adequate margins to accommodate the previously undetermined dynamic effects.

Based on our review of the original static methods employed and the confirmatory evidence obtained from the recent dynamic analyses of the most vulnerable systems, we have concluded that the design of the Class I (Seismic) piping systems in Indian Point Unit 2 is acceptable.

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*See Section 6.1 for definition of Class I structures, systems, and components.

5.5 Inservice Inspection

An inservice inspection program for the reactor coolant system is included in the Technical Specifications. This program follows Section XI of the ASME Code, Rules for Inservice Inspection of the Reactor Coolant System, as closely as practical. The design of the primary system including the capability to remove insulation at selected areas provides an acceptable degree of access for inspection purposes. The applicant also intends to conduct periodic inservice inspections of the primary pump motor flywheels.

The applicant will review the inservice inspection program with us after five years of reactor operation. It may then be modified based on experience gained during these five years. At that time, we will also require the applicant to perform such inspections of components outside the reactor coolant pressure boundary as deemed necessary to provide continuing assurance of structural integrity.

5.6 Missile Protection

We have reviewed the applicant's primary system layout within the containment in terms of the protection afforded the containment liner and Class I (seismic) systems inside the containment from missiles that might be generated as a result of a primary system failure. We have concluded that adequate protection from potential missiles is provided by the system arrangement and surrounding thick circumferential concrete walls and the concrete floors.

The primary pump motor flywheels installed in Indian Point Unit 2 are the same as those in use in other plants. The flywheels are the standard Westinghouse design, fabricated of A 533B steel. On the basis of the use of high grade material, extensive quality control measures, special manufacturing procedures and preservice and inservice surveillance requirements, we have concluded that assurance has been provided that the integrity of the flywheels will be maintained.

5.7 Leak Detection

The reactor coolant pressure boundary leak detection systems for this plant are similar to those we have reviewed and found acceptable for other plants using a Westinghouse nuclear steam supply system. The systems are based upon air particulate monitoring, radiogas monitoring, humidity detection, and containment sump level monitoring. These systems provide an array of instrumentation that is sensitive, redundant, and diverse and that has adequate alarm features. The sensitivity of these systems is consistent with their primary purpose of detecting any leak in the primary system pressure boundary which could be indicative of incipient failure. The Technical Specifications require that two reactor coolant leak detection systems of different principles shall be in operation when the reactor is operated at power. We conclude that the leak detection systems for Indian Point Unit 2 are acceptable.

5.8 Fuel Failure Detection

The fuel element failure detection system will measure delayed neutron activity in one hot leg of the reactor coolant system. The monitor is connected in series with a delay coil to allow a decay time for N^{16} gamma activity (half life of 7.1 seconds) of about 60 seconds before the coolant reaches the detector. This delay reduces gamma ray background and facilitates detector sensitivity. An alarm signal is provided for the channel. We conclude that this system which is inherently faster in response than previous systems reviewed for other reactors is acceptable.

5.9 Vibration Monitoring and Loose Parts Detection

The major core and core support components have been analyzed to provide assurance that they are not vulnerable to vibratory excitation. Vibration analyses for the core support barrel considered inlet flow impingement and turbulent flow. Natural frequency calculations were made to assure that there would be no deleterious response to known excitations such as pump blade passing and driven frequencies. Fuel bundle response to anticipated driving forces has been calculated and determined by tests in the Westinghouse Reactor Evaluation Center.

The vibration monitoring system to be used for the preoperational test program on Indian Point Unit 2 will consist of mechanical gauges to measure gross relative motion between the thermal shield and core barrel, strain gauges on selected guide tubes, and accelerometers on the upper core plate. We have concluded that the vibration design analyses and the preoperational test program are acceptable.

In the course of our review of the Indian Point Unit 2 application, it has been noted that techniques for the analysis of neutron noise spectra and accelerometer measurements on the lower heads of primary system vessels might be developed to provide a useful method for inservice monitoring of reactor coolant systems to detect changes in the vibration of reactor components or the presence of loose parts. The applicant has stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefullness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

5.10 Conclusion

Based on our review of (1) the codes and standards used for design, (2) the fabrication and inspection procedures, (3) the inservice inspection program, (4) the provisions for missile protection and leak detection, (5) the provision for fuel failure detection, and (6) the provisions for preoperational vibration testing and the developmental effort for inservice monitoring to detect vibrations and loose parts, we have concluded that the design and inspection procedures for the reactor coolant system for the Indian Point Unit 2 are acceptable.

6.0 CONTAINMENT AND CLASS I (SEISMIC) STRUCTURES

6.1 General Structural Design

The applicant has categorized as Class I (seismic) those structures (e.g., containment structure and primary auxiliary building), and those systems and components (e.g., reactor vessel and internals, emergency core cooling system), whose failure could cause a significant release of radioactivity or that are vital to the safe shutdown of the facility and the removal of decay heat. We have reviewed the applicant's classification of structures, systems, and components and conclude that they have been classified appropriately.

The Class I (seismic) structures at Indian Point Unit 2 are the containment structure, the primary auxiliary building, the control room building, the fuel storage pool, the diesel generator building, and the intake structure and service water screenwell. The major portion of the primary auxiliary building, the fuel storage pool, and the intake structure are of reinforced concrete construction. The control room building, the diesel generator building, the fuel storage building and the non-Class I portions of the primary auxiliary building are constructed of steel framing with composite metal panel siding.

The environmental conditions that were considered in the structural design include the operating basis earthquake (OBE), the design basis earthquake (DBE), the flooding and wind due to the probable maximum hurricane, and the flooding due to the probable maximum flood. We have concluded that these conditions were used for the design in an acceptable manner.

6.2 Structural Design and Analysis

The Indian Point Unit 2 primary containment has a free volume of 2.6×10^6 cubic feet and a design pressure of 47 psig. The containment structure is a right cylinder (thickness 4.5 ft) with hemispherical dome (thickness 3.5 ft) mounted on a flat (thickness 9 ft) base mat. The reinforced concrete is lined with 1/4 inch minimum thickness welded ASTM A442 grade 60 firebox quality carbon steel plate. The reinforcing bars conform to ASTM A432 specifications. The reinforcing in the cylinder wall is placed in horizontal and vertical directions with added diagonal tangential reinforcing for earthquake resistance. The reinforcing bars conform to ASTM A432 specifications. Cadweld splices are used in 14S and 18S bars.

We have evaluated the pressure transients that might occur in the containment in the event of a loss-of-coolant accident assuming various sizes of primary coolant system breaks. For the range of postulated break sizes up to and including the double-ended severance of the largest reactor coolant pipe, the largest calculated peak containment pressure is 40 psig. The design pressure of the containment exceeds the calculated peak pressure by more than 10% and is acceptable.

The containment is designed to remain within the elastic range for the 0.10g OBE concurrent with the accident and other applicable loads. It is also designed to withstand the 0.15g DBE concurrent with the accident without loss of function.

We and our seismic design consultant, Nathan M. Newmark, are in agreement with the loading combinations and allowable stresses used by the applicant. Stress and strain limits conform to the requirements of ACI 318-63, Part IV-B. The ACI load factors have been replaced by factors suitable for concrete containment structures.

Based on our review of the design of the containment structure and its capability to withstand the predicted pressures from potential accidents, we conclude that the structural design aspects of the containment are acceptable.

In evaluating the capability of the Class I (seismic) structures, systems, and components, to withstand the dynamic loads due to seismic events, our seismic design consultant, Nathan M. Newmark Consultant Engineering Services, considered the geology and nature of the bedrock, design loads and load combinations, the seismic design parameters, and methods of analysis. On the basis of our review and that of our seismic design consultant, we conclude that the Class I (seismic) structures, systems, and components of Indian Point Unit 2 are designed to accommodate all applicable loads and are acceptable. The report of our seismic design consultant is attached as Appendix G.

During our review we noted a limited number of cases where failure of non-Class I (seismic) structures could potentially endanger Class I (seismic) structures and equipment. These included the Indian Point Unit 1 superheater stack and superheater building, the turbine building, and the fuel storage building. In response to our concern, the applicant performed analyses of these structures using a multi-degree of freedom modal dynamic analysis method, to determine the modifications needed to assure that gross structural collapse of these structures would not occur in the event of a DBE. As a result of these analyses, additional seismic reinforcement is being provided for both the superheater building and the turbine building and the Indian Point Unit 1 superheater stack is to be reduced in height by 80 feet. The truncation of the stack is to be accomplished at a convenient time in the next three years and prior to operation of Indian Point Unit 3. We and our seismic design consultant have reviewed the material submitted by the applicant and conclude that the dynamic analyses performed, and the design modifications proposed, are acceptable.

We have reviewed the as-built wind resistance of Class I structures at the Indian Point Unit 2 facility. Analysis indicates that both the containment and reinforced concrete portions of the primary auxiliary building and intake structure can sustain winds in the range of 300 miles per hour. The control building and diesel generator building which are constructed of structural steel with composite metal panel siding, are estimated by the applicant to be capable of sustaining wind loads of up to 160 miles per hour.

Some natural protection from high winds is afforded the control room building and diesel generator building since they are protected by the turbine building to the west, the Indian Point Unit 1 turbine building, superheater building, and containment to the south, the rising hillside to the east, and the containment and rising hillside to the north.

The wind resistance of the Indian Point Unit 1 superheater stack was also considered with respect to preserving the integrity of Indian Point Unit 2. A reduction in stack height of 80 feet coupled with the additional seismic reinforcement of the superheater building (see discussion above) will enable the stack to resist winds with speeds greater than 300 miles per hour.

On the basis of the very low probability for wind speeds greater than 100 miles per hour at the Indian Point site and on the basis of the wind resistance of the Class I (seismic) structures as discussed above, we conclude that Indian Point Unit 2 is adequately protected against high winds.

6.3 Testing and Surveillance

Strength and leakage tests of the containment building will be performed after construction is completed. A 115% overpressure strength test at 54 psig will be conducted and leakage tests will be made at pressures up to 47 psig. As noted in Section 7.3 of this evaluation, pressurized test channels are provided at all liner seams for long-term surveillance. No permanent instrumentation system is designed to the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Criteria No. 279 for protection systems. The Technical Specifications require periodic testing of the overspeed devices to assure operability. We conclude that the applicant has made appropriate provisions to reduce the probability of a destructive turbine missile from being generated and affecting Class I (seismic) items.

The Indian Point Unit 2 reactor vessel cavity is designed to protect the containment against missiles that might be produced by postulated failure of the reactor vessel. Failure of the reactor vessel would result in fluid jet-reaction forces in the cavity wall adjacent to the vessel split or crack as well as stress in the cavity wall from a rise in cavity pressure, both of which would result from coolant blowdown. Also reaction forces in the cavity wall and floor might be produced by the impact of missiles generated by pressure vessel failure. By the use of extensive steel reinforcing, the concrete cavity has been designed to resist both fluid jet and missile impact forces that could result from pressure vessel failure by either longitudinal splitting or various modes of circumferential cracking. The cavity is also designed to sustain a fluid pressure rise to 1000 pounds per square inch. We have reviewed the applicant's analysis and conclude that the cavity as designed provides adequate protection for the containment liner against missiles that might result from a postulated pressure vessel failure.

7.0 ENGINEERED SAFETY FEATURES

7.1 Emergency Core Cooling System

The principal equipment of the emergency core cooling system consists of (1) three 50% capacity high pressure safety injection pumps, (2) two 100% capacity residual heat removal pumps for low pressure injection and external recirculation, (3) two 100% capacity recirculation pumps for recirculation internal to the containment, (4) one 100% capacity boron injection tank, and (5) four 33-1/3% capacity accumulators. This system provides redundant capability to inject borated cooling water rapidly into the core in the event of a loss-of-coolant accident and to maintain coolant above the level of the core for an indefinite period following the accident.

The applicant's evaluation of the performance of these systems is based on detailed analyses of (1) the hydraulic behavior of the primary coolant system during and subsequent to a loss-of-coolant accident, and (2) the thermal response of the core during the same period. The analytical methods used to predict the hydraulic behavior of the primary coolant system during a loss-of-coolant accident have been improved significantly during the construction period for Indian Point Unit 2. The original analysis presented in Volume 4 of the FFDSAR was performed with the FLASH-1 hydraulics computer program. This program is limited to a three-node representation of the coolant system. Subsequent to the analysis performed with FLASH-1, Westinghouse developed a new multi-node hydraulics program called SATAN. Using SATAN the coolant system can be represented with as many as 96 nodes. The SATAN calculations provide considerable detail in the system analysis and increased insight into system performance.

At our request, the applicant reevaluated the performance of the emergency core cooling system during a loss-of-coolant accident using the SATAN multi-node hydraulics code. The applicant's analysis is based on the license application power rating of 2758 MWt. For the case of an accident initiated by a double-ended break in the cold leg primary coolant piping, a maximum fuel element clad temperature of 2015°F was predicted. The applicant's investigation of the emergency core cooling system performance for a range of break sizes and locations indicates that the resultant peak temperatures for any other break will be less than those predicted for the double-ended cold leg break. On the basis of our review of the analytical techniques used in this analysis and our experience with similar analytical techniques, we conclude that there is reasonable assurance that the results obtained with these techniques provide a conservative estimate of the performance of the system in the event of a loss-of-coolant accident at Indian Point Unit 2.

We conclude that the emergency core cooling system will (1) limit the peak clad temperature to well below the clad melting temperature, (2) limit the fuel clad water reaction to less than 1% of the total clad mass, (3) terminate the clad temperature transient before the geometry necessary for cooling is lost and before the clad is so embrittled as to fail upon quenching and (4) reduce the core temperature and then maintain core and coolant temperature levels in a subcooled condition until accident recovery operations can be accomplished.

In summary, we conclude that the emergency core cooling system is acceptable and will provide adequate protection for any loss-of-coolant accident.

The emergency core cooling system design as presently installed at Indian Point Unit 2 was reviewed by the Division of Reactor Licensing during 1967, subsequent to the issuance of the construction permit on October 14, 1966. This system represented a complete redesign, a considerable increase in flow capability, and enhanced performance when compared to the system reviewed for the construction permit. On the basis that the very significantly improved performance of the redesigned emergency core cooling system provides additional assurance for limiting clad temperatures and maintaining a coolable core we concurred with the applicant's decision to remove the reactor pit crucible from the facility design.

7.2 Containment Spray and Cooling Systems

Two independent heat removal systems are provided to control the containment pressure and temperature following a loss-of-coolant accident. Each system, acting alone at its rated capacity, will prevent over-pressurization of the containment structure. The two systems are the containment spray system and the fan cooling system. The design of each is substantially the same as the design of systems provided at the Ginna plant and other licensed plants.

The containment spray system consists of two 50% capacity spray pumps and is sized to limit the containment post-accident pressure to below design pressure. Sodium hydroxide and boric acid are used as additives to the spray solution to remove radioactive iodine which might be present in the containment after an accident. We have reviewed the use of these chemical spray additives in terms of their iodine removal capabilities, and in addition have evaluated the chemical compatibility of the spray solution with other reactor components. As a result of our review, we conclude that the spray system is adequately sized to cool the containment, that the alkaline spray solution will reduce the iodine concentration in the containment atmosphere, and that corrosion of other materials used in the containment does not introduce a safety problem.

The containment fan cooling system provides complete redundancy to the containment spray system for heat removal from the containment atmosphere during post-accident conditions. Five 20% capacity fan coolers are provided. Since the fan coolers are located within containment, they must be capable of operating in the post-accident environment. Westinghouse has conducted an environmental test program to demonstrate this capability. Our evaluation of these tests, including the heat removal capability of the heat exchangers, and environmental and radiation testing of the fan cooler motors, valve motor operators and electric cabling indicates that these components will function satisfactorily in the accident environment. An iodine-impregnated charcoal filter system has been included with the fan cooler system to remove organic iodine from the post loss-of-coolant containment atmosphere. The charcoal beds are preceded by demisters and high efficiency particulate air (HEPA) filters.

We have evaluated the inorganic and organic iodine removal capability of the charcoal beds on the basis of tests with steam - air mixtures at 100% relative humidity following prolonged flooding of the bed. We conclude that inorganic and organic iodine removal efficiencies of 90% and 10% per pass, respectively, are conservative values that are justified by the available information.

In summary, we have reviewed the containment spray and fan cooling systems in terms of (1) capability to control the containment temperature, (2) capability to remove inorganic and organic iodine,

(3) system and component redundancy, and (4) capability to function in the post-accident containment environment. We conclude that there is reasonable assurance that these systems will operate as proposed subsequent to a loss-of-coolant accident.

7.3 Containment Isolation Systems

In addition to the usual capability of isolating all lines leading to and from the containment, the Indian Point Unit 2 containment is provided with additional systems to minimize the potential leakage of fission products subsequent to an accident. A containment penetration and weld-channel pressurization system provides for continuous pressurization of zones enclosing containment penetrations and the welds in the containment liner. The system continuously maintains an overpressure of clean, dry air that is in excess of the containment design pressure. Pressurized zones include each piping penetration, each electrical penetration, double gasketed spaces on the personnel and equipment hatches, and the channels over weld seams of the containment liner. The air pressure is maintained by the instrument air compressors with backup from the plant air compressors and from a standby source of nitrogen cylinders. Pressure indication and alarm instrumentation is provided locally and in the control room to assure that loss of pressure will be detected and corrected.

In addition, an isolation seal water system has been provided to assure containment isolation by (1) injecting seal water between the seats and stem packing of the globe and double disc isolation valves used on larger lines, and (2) injecting seal water directly into the line between the closed diaphragm valves used in the smaller lines penetrating containment. Seal water injection is provided for all lines connected to the reactor coolant system and for lines that may be exposed to the containment atmosphere subsequent to an accident. Although the use of the seal water system following a loss-of-coolant accident provides an additional means of reducing leakage, we have not considered the effect of this system in determining the offsite radiological consequences.

We have concluded that the capability provided for isolating the containment is acceptable.

7.4 Post-Accident Hydrogen Control System

In the event of a loss-of-coolant accident, radiation from the core and from escaped fission products will dissociate some of the cooling water into gaseous hydrogen and oxygen. Continued evolution of hydrogen would increase the concentration in the containment to a point where ignition could occur and thus provide an additional energy source.

Redundant flame recombiner units are installed within the Indian Point Unit 2 containment. Each unit has the design capability to prevent the ambient containment hydrogen concentration from exceeding two percent by volume. The units are designed to function, following the loss-of-coolant accident in a containment pressure environment of 1 to 5 psig. Each recombiner system consists of (1) a flame recombiner unit located within containment, (2) a control panel located outside of containment, and (3) a hydrogen gas stand located outside of containment. On the basis of (1) our detailed review of the design of the system and its controls, (2) satisfactory performance testing of the device, and (3) satisfactory environmental testing of those portions of the recombiner system installed within the containment, we conclude that there is reasonable assurance that the recombiner system will perform its intended post-accident function.

In addition, the applicant will provide the capability for purging the containment atmosphere through appropriate filters as an alternate backup means of hydrogen control. The containment penetrations to be used for this system are installed. The design and installation of the equipment required will be performed during the first two years of operation at power.

8.0 INSTRUMENTATION, CONTROL, AND POWER SYSTEMS

8.1 Reactor Protection and Control System

The reactor protection system instrumentation for Indian Point Unit 2 is the same as that installed at the Ginna plant. The adequacy of the protection system instrumentation was evaluated by comparison with the Commission's proposed general design criteria published on July 11, 1967, and the proposed IEEE criteria for nuclear power plant protection system (IEEE-279 Code), dated August 28, 1968. The basic design has been reviewed extensively in the past and we conclude that the design for Indian Point 2 is acceptable.

During our review we considered the adequacy of reactor protection for operation with less than four coolant loops in service. When operating with one of the primary loops out of service the reactor is normally automatically limited to 60% of full power. However by manual adjustment of several protection system set points in a manner consistent with the Technical Specifications adequate reactor protection can be provided for operation up to 75% of full power.

We have reviewed the applicant's analysis of the seismic response of the protection system instrumentation and associated electrical equipment and find that adequate testing has been performed on the nuclear instrumentation, switch gear, and process system instrumentation.

In connection with our review of potential common mode failures we have recently considered the need for means of preventing common failure modes from negating scram action and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has been responsive to our request for information and has provided the results of analyses which indicate that the consequences of such transients are tolerable for the existing Indian Point Unit 2 design at a power level of 2758 MWt. Although additional study is required of this general question, we conclude that it is acceptable for the Indian Point Unit 2 reactor to operate at a power level of 2758 MWt while final resolution of this matter is made on a reasonable time scale.

8.2 Initiation and Control of Engineered Safety Features

The instrumentation for initiation and control of engineered safety features for the Indian Point Unit 2 is the same as that installed at the Ginna plant. This basic design has been reviewed extensively in the past and we consider it to be acceptable.

We have reviewed the capability for testing engineered safety feature circuits during reactor operation. Resistance tests will be used for routine determinations of the operability of the master and slave relay coils. The circuits upstream of these relays can be partially tested during operation. During plant shutdown, circuits can be tested completely by coincident tripping of instrument channels and a consequent operation of the master and slave relays in the entire downstream initiating system. We have concluded that this testing capability is acceptable for Indian Point Unit 2.

8.3 Off-Site Power

Two 138 kilovolt (kV) lines connect the Buchanan switchyard to the Millwood switching station, which in turn is connected to the Consolidated Edison grid and the Niagara Mohawk and Connecticut Light and Power systems. Two additional 138 kV lines, using a separate route from the first two lines, connect the switchyard to the Orange and Rockland tie.

The applicant stated that an analysis of the transmission system has indicated that the system is stable for the loss of any generating unit including Indian Point Unit 2.

A single 138 kV line connects the Buchanan switchyard to Indian Point Unit 2. In addition, three 13 kV lines connect the switchyard to Indian Point Unit 1. Three 138/13 kV transformers in the switchyard feed these three 13 kV lines. While the 138 kV system is the normal supply for the auxiliary load associated with plant engineered safety features, one of the three Indian Point Unit 1 13 kV lines is available to provide power via automatic switching to Indian Point Unit 2 through a 13/6.9 kV transformer. By switching circuit breakers in Indian Point Unit 1, the other two 13 kV lines can also be made available to provide power to Indian Point Unit 2. As the 13/6.9 kV supply is not capable of carrying the total plant auxiliary load for Indian Point Unit 2, the main coolant pumps and the circulating water pumps must be tripped off before the supplies are switched.

We conclude that the off-site power supply provides an adequate source of power for the engineered safety features and safe shutdown loads.

8.4 Onsite Power

Onsite power is supplied by three independent diesel generator sets connected in a separate bus configuration such that there is no automatic closure of tie breakers between the three buses to which the generators are connected. The redundant engineered safety feature (ESF) loads are arranged on the three separate buses such that failure of a single bus will not prevent the required ESF performance under accident conditions. The design engineered safety feature and safe shutdown loads per diesel generator are 1813, 2210, and 2353 HP for the first one-half hour following a loss-of-coolant accident. The loads are then changed to 2438, 2235, and 2043 HP for the recirculation phase of the emergency core cooling system operation. On the basis of our evaluation, we have determined that the appropriate diesel generator ratings are 2200 HP continuous, and 2460 HP for 2,000 hours. We note that some of the estimated emergency loads are above the continuous rating of the machines, but below the 2,000 hour ratings. We consider that this margin is acceptable for Indian Point Unit 2.

Each diesel generator is started automatically upon initiation of emergency core cooling system operation or upon under-voltage on its corresponding 480-volt emergency bus. The generators are housed in a separate Class 1 (seismic) structure. On-site diesel fuel storage capacity provides a minimum of seven days operation at the required safety feature loads. These design and operating features are acceptable for Indian Point Unit 2.

Our review of the ac auxiliary power system has disclosed that there is adequate capacity and an adequate degree of physical and electrical separation of redundant features. The 125 volt dc system consists of two individually housed batteries. The dc system is divided into two buses with a battery and battery charger for each bus. Each of the two station batteries has been sized to carry its expected loads for a period of two hours following a plant trip at a loss of all ac power.

We conclude that the onsite emergency power system is acceptable.

8.5 Cable Installation

We have reviewed the applicant's cable installation relative to the preservation of the independence of redundant channels by means of separation, and relative to the prevention of cable fires through proper cable rating and tray loading. This has been performed by reviewing the cable installation criteria and method of layout design and by field inspection of electrical cable installation during construction.

A single electrical tunnel carries the electrical cables from the electrical penetration area of the containment to the control building. This tunnel carries all of the electrical cables except the power cables for the reactor coolant pumps, the pressurizer heater cables, and the control rod power cables. The cables in the tunnel are arrayed on either side of a three-foot aisle in trays or ladders. Separation is provided for in the form of distance, metal separators, or transite barriers. The electrical tunnel does not contain any spliced cable connections. Therefore, the probability of a fire is reduced. Further, a fire detection system and an automatically operated water spray system are provided in the tunnel. Tunnel cooling is provided for by redundant cooling fans. On the basis of adequate separation within the tunnel, a minimum number of heat producing cables and features, redundant cooling systems, and fire detection and spray systems we conclude that the single electrical tunnel is acceptable.

Sixty electrical penetrations are provided in a single electrical penetration area to provide for entry of signal, control, and power cables into the containment. The penetrations are located on three-foot centers, both horizontally and vertically, and are of the hermetically sealed type. As a result of our review, fire barriers in the form of transite sheets were added to separate the power cable penetration from the instrument and control cable penetrations. In addition, as a result of our review certain modifications were made to the cabling in the penetration area, including shortening of cable runs and elimination of cable loops. The segregation of power cables and the shortening of the cable runs reduces the probability of failure by fire and on this basis, we consider the single electrical penetration area acceptable for Indian Point Unit 2.

The applicant has performed a design audit to verify the separation of redundant engineered safety feature power and control electrical cabling. A design review of instrument cabling was also performed on a sample basis.

On the basis of our review of cable installation at Indian Point Unit 2, we conclude that the resulting cable layout, as installed, is acceptable.

8.6 Environmental Testing

Westinghouse has conducted an environmental test program for the instrumentation and controls that are located inside containment and that must function in the environment following a loss-of-coolant accident. We have reviewed the results of this testing program and conclude that the essential instrumentation and controls will function properly in the accident environment.

9.0 RADIOACTIVE WASTE CONTROL

Liquid and gaseous waste handling facilities are designed to process waste fluids generated by the plant so that discharge of liquid and gaseous effluents to the environment will be minimized. Liquid waste is processed both by direct removal of radioactive material with ion exchange resins and by evaporative separation. Using these methods the volume of radioactive waste will be greatly concentrated and the purified liquid streams will either be reused or discharged. Small quantities of radioactive liquid waste will be released routinely to the condenser circulating water discharge canal common to all three units where the waste will be diluted and discharged to the Hudson River.

The limits on routine radwaste releases from the three units that are planned for operation at the Indian Point site will require that the combined releases from the three units when added together be within the limits specified in 10 CFR Part 20. This requirement is stated in Section 3.9 of the Technical Specifications for both liquid and gaseous effluents.

The liquid effluent releases from the three nuclear facilities will be discharged from a common discharge canal into the Hudson River. The nearest sources of public drinking water supplies from the Hudson River are located at Chelsea, New York (backup water supply for New York City) and at the Castle Point Veterans Hospital, 22 and 20.5 miles upstream of the Indian Point site, respectively.

During dry periods with low fresh water river flow, tidal action could carry the radioactivity discharge into the river at the Indian Point site upstream to these river water intake points. Conservative analyses made by the applicant indicate that the concentration of radionuclides at these public water intake points would be less than 1% of the concentration of radionuclides being discharged into the river at Indian Point. Since the releases at the site will be less than the limits of 10 CFR Part 20 (and are expected to be less than 10% of the 10 CFR Part 20 limits, based on past experience with Indian Point Unit 1 and other pressurized water reactor plants), the radioactivity levels at these intakes due to the discharges at Indian Point will not be significant.

Gaseous wastes containing some radioactivity are stored in one of four gas decay tanks. One gas tank is utilized for filling, one for holdup for a 45-day decay period, one for discharging to the atmosphere, and one is held in reserve. Disposal of gaseous wastes from Indian Point Unit 2 is by discharge through the plant vent.

The routine gaseous radioactivity releases from the three nuclear facilities will be from three different vents. The combined release of gaseous waste containing radioactivity from these three sources will be limited by the Technical Specifications such that annual average concentrations at the minimum exclusion distance will not exceed the limits of 10 CFR Part 20, Appendix B, of the Commission's regulations. For gaseous halogens and particulates with half-lives greater than eight days, the applicable limits of the Technical Specifications are less than 1% of the limits given in 10 CFR Part 20. The Technical Specifications also require that the maximum release rate of gaseous waste not exceed the annual average limit.

Based on our review we conclude that the means provided by the applicant for the disposal of radioactive waste are substantially the same as those we have approved for other facilities and are acceptable. We also conclude that acceptable means are provided and will be used to keep the release of radioactivity from the plant within ranges that we consider to be as low as practicable.

10.0 AUXILIARY SYSTEMS

The auxiliary systems necessary to assure safe plant shutdown include (1) the chemical and volume control system, (2) the residual heat removal system, (3) the component cooling system, and (4) the service water system. The systems necessary to assure adequate cooling for spent fuel include (1) the spent fuel pool cooling system, (2) the fuel handling system, and (3) the service water system. The designs for these systems are substantially the same as those we reviewed and found acceptable for the Ginna plant.

10.1 Chemical and Volume Control System

The chemical and volume control system (1) adjusts the concentration of boric acid for reactivity control, (2) maintains the proper reactor coolant inventory and water quality for corrosion control, and (3) provides the required seal water flow to the reactor coolant pumps. The amount of boric acid to be added to the core for reactivity control is determined by the operator. The addition of unborated water as a result of operator error could result in an unintentional dilution during refueling, reactor startup, and power operation. The applicant's analysis indicated that because of the slow rate of dilution there is ample time for the operator to become aware of the dilution and to take corrective action. The applicant is actively participating in the development of a device for continuous monitoring of the reactor coolant boron concentration and will evaluate the feasibility of installing such a monitor when developed.

Our review of the chemical and volume control system emphasized those portions involved in routine and emergency injection of concentrated boric acid. We conclude that the design is acceptable.

10.2 Auxiliary Cooling Systems

Subsystems for auxiliary cooling are the component cooling system, the residual heat removal loop, the spent fuel pool cooling loop, and the service water system. The piping for these three systems is designed to the ANSI B31.1 Code for Pressure Piping.

These systems are equivalent in purpose and design to those of other recently licensed plants. On the basis of our review of this plant and others using the similar systems, we have concluded that these systems are acceptable.

10.3 Spent Fuel Storage

The fuel handling system is designed to transfer spent fuel to the storage pool and to provide storage for new fuel. The spent fuel storage facility is basically the same in capacity and design as those used in previously licensed pressurized water reactor plants. The fuel pool is sized to accommodate spent fuel from 1-1/3 core loadings.

As in other designs, mechanical stops will be incorporated in the crane to restrict motion of the spent fuel cask to its assigned area, adjacent to one side of the fuel storage pool. In addition, the spent fuel racks in the area adjacent to the fuel cask storage location would be used only in the event that a complete core is unloaded and one-third of a core from a previous unloading is already in storage.

The pool floor is located below grade level and founded on solid rock. Structural damage from a dropped fuel cask would not result in a rapid loss of water from the pool. Makeup water can be supplied from the demineralizer water supply at a flow rate of 150 gpm. Additional water can be provided in an emergency by the use of temporary hookups to other sources.

As a consequence of our evaluation of the potential consequences of a postulated fuel handling accident, the applicant has agreed to provide charcoal filters in the refueling building to reduce the calculated offsite doses that might result in the event of a fuel handling accident in the refueling building. The installation of the filters will be completed during the first year of full power operation.

We conclude that the designs of the spent fuel storage pool and the fuel handling system are acceptable.

11.0 ANALYSES OF RADIOLOGICAL CONSEQUENCES FROM DESIGN BASIS ACCIDENTS

11.1 General

In order to assess the safety margins of the plant design, a number of operating transients were considered by the applicant, including rod withdrawal during startup and at power, moderator dilution, loss of coolant flow, loss of electrical load, and loss of ac power. The reactor control and protection system is designed so that corrective action is taken automatically to cope with any of these transients. Based on our evaluation of the information submitted by the applicant and our evaluations of other PWR designs at the operating license stage, we conclude that the Indian Point Unit No. 2 control and protection system design is such that these transients can be terminated without damage to the core or to the reactor coolant boundary, and with no offsite radiological consequences.

The applicant and we have evaluated the consequences of potential accidents, including a control rod ejection accident, an accident involving rupture of a gas decay tank, a steamline break accident, a steam generator tube rupture accident, a loss-of-coolant accident, and a refueling accident.

The calculated offsite radiological doses that might result from the control rod ejection accident, and the accident involving rupture of a gas decay tank are well within the 10 CFR Part 100 guidelines.

The consequences of the steamline break and the steam generator tube rupture accidents can be controlled by limiting the permissible concentrations of radioactivity in the primary and secondary coolant systems. The Technical Specifications for the Indian Point Unit No. 2 facility limit the primary and secondary coolant activity concentrations such that the potential 2-hour doses at the exclusion radius that we calculate for these accidents do not exceed 1.5 Rem to the thyroid or 0.5 Rem to the whole body.

Our evaluations of the loss-of-coolant accident and the refueling accident are discussed in the following sections.

11.2 Loss-of-Coolant Accident

The design basis loss of coolant accident (LOCA) for the Indian Point Unit No. 2 plant is similar to that evaluated for other PWR plants in that a double-ended break in the largest pipe of the reactor coolant system is assumed.

Although the basis for the design of the emergency core cooling system is to limit fission product release from the fuel, in our conservative calculation of the consequences of the LOCA we have assumed that the accident results in the release of the following percentages of the total core fission product inventory from the core: 100% of the noble gases, 50% of the halogens, and 1% of the solids. In addition, 50% of the halogens that are released from the core is assumed to plate out onto internal surfaces of the containment building or onto internal components and is not available for leakage. We assume that 10% of the iodine available for leakage from the containment is in the form of organic iodide, and that 5% is in the form of particulate iodine. The reactor is assumed to have been operating at a power of 3217 MWt prior to the accident. The primary containment is assumed to leak at a constant rate of 0.1 percent of the containment volume per day for the first day and 0.05 percent per day thereafter. We evaluated the iodine removal capability of the sodium hydroxide containment spray system and assumed an inorganic iodine removal constant of 4.5 per hour for the spray system. We evaluated the iodine removal capability of the iodine impregnated charcoal filter system and assumed a removal constant of 0.49 per hour for inorganic iodine and a removal constant of 0.048 per hour for organic iodine. Iodine particulates are assumed to be removed by the high efficiency particulate air filters. The inhalation rate of a person offsite is assumed to be 3.5×10^{-4} cubic meters per second.

For the calculation of the two-hour dose at the site boundary we used an atmospheric dispersion factor corresponding to Pasquill Type "F" stability, with a 1 meter per second wind speed and an appropriate building wake effect. We calculated the potential doses at the site boundary for this 2 hour period to be 180 Rem to the thyroid and 4 Rem to the whole body. At the low population zone boundary our calculated potential doses for a 30-day period are 270 Rem to the thyroid and 7 Rem to the whole body.

In evaluating the above doses, no credit was given for the isolation valve seal water injection system, the penetration pressurization system, or the weld channel pressurization system. Operation of these systems, which interpose a high gas pressure or seal water area between the containment and the outside atmosphere at all points where leakage might occur, should significantly reduce the leakage rate from the containment, and, thus reduce the doses following an accident. These systems are well designed and tested, and should be available in the event of an accident (see Section 7.3). We did not consider the effect of these systems in our dose calculations because it is inherently difficult to accurately measure leakage rates of less than 0.1% per day by current testing methods.

The control room for Indian Point Unit No. 2 was not designed to meet the requirements we have imposed in more recent construction permit reviews, that the dose for the course of the accident to occupants of the control room be limited to 5 Rem to the whole body and 30 Rem to the thyroid. In order to provide additional protection to the control room occupants in the event of a loss-of-coolant accident, the applicant has equipped the control room with protective clothing and self-contained air respirators for the operators. In view of these provisions, we have concluded that the control room, as constructed, is acceptable in this regard.

11.3 Fuel Handling Accident

We have evaluated the potential consequences of a fuel handling accident, in which it is postulated that a fuel assembly is dropped in the spent fuel pool or transfer canal. We assumed that: (1) all 204 rods in the dropped bundle are damaged, (2) the accident occurs 90 hours after shutdown of the core from which the dropped bundle has been removed, (3) 20% of the noble gases and 10% of the iodine in the dropped fuel bundle are released to the refueling water and the dropped fuel bundle has been removed from a region of the core which has been generating 1.43 times the average core power, (4) 90% of the released iodine is retained in the refueling water, (5) the fission products released from the pool are discharged to the atmosphere by the building recirculation system through charcoal filters with an iodine removal efficiency of 90%, and (6) the same meteorological conditions exist as were assumed for the loss-of-coolant accident. The resultant calculated doses at the site boundary are 146 Rem to the thyroid and less than 4 Rem to the whole body.

11.4 Conclusions

We have calculated offsite doses for the design basis accidents that have the greatest potential for offsite consequences using assumptions consistent with those we have used in previous safety reviews of PWR plants and have found the resulting calculated doses to be less than the guideline values of 10 CFR Part 100.

12.0 CONDUCT OF OPERATIONS

12.1 Technical Qualifications

The Indian Point Unit 2 facility was designed and is being built by Westinghouse as prime contractor for the applicant. Preoperational testing of equipment and systems at the site and initial plant operation will be performed by Consolidated Edison personnel under the technical direction of Westinghouse. The applicant's experience in the power production field is largely with thermal power plants. However, the applicant has operated Indian Point Unit 1, a 615 megawatt (thermal) pressurized water reactor plant with an oil fired superheater, since August 1962. In addition, the applicant has the Indian Point Unit 3 under construction at the Indian Point site and is actively considering the installation of other nuclear power plants at other sites. Our review of the applicant's organization indicates that the competence of its engineering staff has continually increased and is consistent with the requirements of its expanded nuclear program.

12.2 Operating Organization and Training

The applicant's organization consists of three main groups under the direction of the general superintendent. These groups are the operations group (with a separate superintendent for each unit), the performance group (with the responsibility for station chemistry, licensed personnel training, and surveillance of station performance), and the health physics group headed by a supervisor engineer for health physics (with the responsibility for station health physics and instrumentation). An assistant superintendent for maintenance, and production engineers (responsible for providing staff support for the operation superintendents) report to the two superintendents for operation. A reactor engineer reports directly to the general superintendent.

The proposed shift complement for the combined operation of Indian Point Unit 1 and Indian Point Unit 2 consists of one general watch foreman licensed as a senior reactor operator (SRO), one watch foreman (SRO) for each unit, one control operator A licensed as a reactor operator (RO) for each unit, one unlicensed control room operator B, shared by both units, one control operator B for Indian Point Unit 1 chemical system building, six operating mechanics (two of whom are assigned to Indian Point Unit 2), one shift chemist, and one shift health physics technician.

The shift composition for Indian Point Unit 2 when Indian Point Unit 1 is shutdown for any reason is the general foreman, one watch foreman, one control operator A and two operating mechanics. In addition, a control room operator B may be available a substantial portion of his time. We conclude that both the dual unit crews and single unit crews as outlined above are acceptable.

Since a large part of the plant staff has had prior nuclear experience, the training program has been fitted to individual needs based on experience, educational background and job responsibilities. The training program includes long- and short-term assignments of key staff personnel to technical institutions and operating reactors, to the Westinghouse offsite operator training school, and to on-site classroom training courses for operators and supervisors conducted by both applicant and Westinghouse personnel. We have reviewed these activities in detail and conclude that the combination of reactor operating experience and formal training obtained by the plant staff has adequately prepared them to perform their operational duties.

As a means for the continuing review and evaluation of plant operational safety, the applicant will expand the responsibilities of the Nuclear Facility Safety Committee currently functioning for Indian Point Unit 1 to include Indian Point Unit 2. The committee, which reports to the Executive Vice President, Central Operations, will have a membership of at least 12 persons, and will have responsibilities to: (1) audit and report upon the adequacy of all procedures used in the operation, maintenance, and environmental monitoring of each nuclear plant; (2) review and report upon the adequacy of all proposed changes in plant facilities and procedures pertaining to operation, maintenance, and environmental monitoring and having safety significance;

(3) review and report upon all proposed changes to the Technical Specifications; (4) conduct unannounced spot inspections of plant monitoring operations; (5) review and report upon any activity, the occurrence or lack of which may affect the safe operation of the nuclear plant; and (6) convene, at the request of the nuclear power generation manager or a nuclear plant general superintendent or chairman or vice chairman of the committee, to review and act upon any matter they may deem necessary.

Westinghouse will participate in the startup and initial operation of the plant and will continue to make available technical support to the Indian Point Unit 2 staff during operation of the facility.

We conclude that the applicant's organization is acceptably staffed and technically qualified to perform its operational duties subject to satisfactory completion of licensing examinations of personnel requiring licenses.

12.3 Emergency Planning

The site emergency plan for the Indian Point site describes the emergency organization and its responsibilities. The scope of the emergency plan includes consideration of local contingencies, site contingencies, general (off-site) contingencies, implementation levels for each contingency, notification channels, the support provided by civil authorities, protective measures for each contingency, communications facilities, and training drills. The applicant has provided an extensive description of the medical support that will be available although it is not incorporated explicitly in the plan. The planned medical support provides for emergency treatment of plant personnel both at the site and at a designated hospital where facilities equipment and medical personnel to handle radiation contaminated injured personnel will be available.

We conclude that the applicant's emergency plan is acceptable for Indian Point Unit 2.

12.4 Industrial Security

The immediate plant area (restricted area), including Indian Point Unit 1 will be enclosed by a fence. Access to the restricted area for all personnel will be through manned gatehouses or locked gates which are under the direct control of the station security forces. Security guards will make routine patrols of all property within the site boundary and outside the restricted area and are required to make hourly reports to the central control room.

The controlled area of Indian Point Unit 2 will include the containment, the fuel storage building, the primary auxiliary building, and the emergency diesel generator building. Normal access to these areas is through the existing security room for Indian Point Unit 1. All other doors and hatches leading into the controlled area will be locked and will be supervised by means of door switches connected to the open door alarm board in the security room, and the category alarm board in the Indian Point Unit 1 central control room. The containment personnel hatch doors have remote indicating lights and annunciators that are located in the control room and that indicate the door operational status.

Offsite applicant employees must identify themselves at the main gate prior to admission to the restricted area, receive approval for entry by the general superintendent or his designated representative, and sign in on an admission sheet. If access into the controlled area is approved, they must be accompanied by a qualified guide.

We conclude that the applicant has taken reasonable measures to provide for the security of the facility.

13.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in an operating license define safety limits and limiting safety system settings, limiting conditions for operation, periodic surveillance requirements, certain design features, and administrative controls for the operating plant. These specifications cannot be changed without prior approval of the AEC. The applicant's initial proposed Technical Specifications, presented in Amendment No. 20, have been modified as a result of our review to describe more definitively the allowable conditions for plant operation. The Technical Specifications as approved by the regulatory staff, may be examined in the Commission's Public Document Room.

Based upon our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits and that means are provided for keeping the release of radioactivity from the plant within ranges that we consider as low as practicable. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features to mitigate the consequences of unlikely accidents will be available.

14.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The ACRS reported on the application for construction of the Indian Point Unit 2 at the proposed site in a letter dated August 16, 1966. The applicant has been responsive to the recommendations made by the ACRS in that letter, and we conclude that the matters raised have been resolved satisfactorily during the design and construction of the Indian Point Unit 2.

The ACRS reported on its review of the application for an operating license for Indian Point Unit 2 in their letter, dated September 23, 1970, attached as Appendix B.

In its letter, the ACRS made several recommendations and noted several items all of which have been considered in the indicated sections of our evaluation. These include: (1) reevaluation of potential flooding at the Indian Point site (Section 3.4), (2) additional seismic reinforcing at the Indian Point Unit No. 1 superheater building and truncation of the superheater stack (Section 6.2), (3) reactor design, power distribution, and control of potential xenon oscillations (Section 4.2), (4) containment design and isolation (Sections 6.2 and 7.3), (5) containment cooling and iodine removal systems (Section 7.2), (6) emergency core cooling system and removal of the reactor pit crucible (Section 7.1), (7) post-accident hydrogen control (Section 7.4),

(8) charcoal filters in the refueling building (Section 10.3), (9) reactor core instrumentation (Section 4.2), (10) reactor protection with only three of four loops in service (Section 8.1), (11) inservice vibration monitoring and loose parts detection (Section 5.9), (12) fuel failure detection (Section 5.9), (13) availability requirements for primary coolant leak detection systems (Section 5.7), (14) pressure vessel fracture toughness (Section 5.2), (15) integrity of high burnup fuel during design transients (Section 4.3), and (16) common mode failure and anticipated transients without reactor scram (Section 8.1).

The ACRS concluded in its letter that if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

15.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and all of the directors and principal officers of the applicant are United States citizens.

The applicant is not owned, dominated or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes, is involved. For these reasons and in the absence of any information to the contrary, we have found that the activity to be performed will not be inimical to the common defense and security.

16.0 FINANCIAL QUALIFICATIONS

The Commission's regulations that relate to the financial data and information required to establish financial qualifications for an applicant for an operating license are 10 CFR Part 50.33(f) and 10 CFR Part 50 Appendix C. The Consolidated Edison Company's application as amended by Amendment No. 21 thereto, and the accompanying certified annual financial statements provided the financial information required by the Commission's regulations.

These submittals contain the estimated operating cost for each of the first five years of operation plus the estimated cost of permanent shutdown and maintenance of the facility in a safe condition. The estimated operating costs are $10.0 million for 1971 (the first year of operation), $14.8 million for 1972, $12 million for 1973, $10.9 million for 1974 and $10.7 million for 1975 (Amendment No. 21). Such costs include the costs of operating and maintenance and fuel. The applicant's estimate of the cost of permanently shutting down the facility and maintaining it in a safe condition is (1) $265,000 for the first year of shutdown and $50,000 for each year thereafter if the reactor core is removed from the vessel, and (2) $240,000 per year if the core is not removed.

We have examined the certified financial statements of the Consolidated Edison Company to determine whether the Company is financially qualified to meet these estimated costs. The information contained in the 1969 financial report indicates that operating revenues for 1969 totaled $1,028.3 million; operating expenses (including taxes) was $830.5 million; the interest on the long-term debt was earned 2.3 times; and the net income for the year was $127.2 million, of which $102.1 million was distributed as dividends to the stockholders, and the remainder of $25.1 million was retained for use in the business. As of December 31, 1969, Company's assets totaled $4,069.6 million, most of which was invested in utility plant ($3,793.3 million), and earnings reinvested in the business were $426.1 million. Financial ratios computed from the 1969 statements indicate a sound financial condition, (e.g., long-term debt to total capitalization--0.52, and to net utility plant--0.52; net plant to capitalization--0.994; the operating ratio--0.81; and the rates of return on common--7.7%; on stockholder's investment--6.9%; and on total investment--4.9%). The record of the Company's operations over the past 5 years reflects that operating revenues increased from $840 million in 1965 to $1,028 million in 1969; net income increased from $111.8 million to $127. million; and net investment in utility plant from $3,170 million to $3,793 million. Moody's Investors Service (August 1969 edition) rates the Company's first mortgage bonds as A (high-medium grade). The Company's current Dun and Bradstreet rating (July 1970) is AaAl.

Our evaluation of the financial data submitted by the applicant, summarized above, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR Part 50.33(f) with respect to the operation of Indian Point Unit 2. A copy of the staff's financial analysis is attached as Appendix H.

17.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licensees for facilities such as power reactors under 10 CFR Part 50.

17.1 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The Consolidated Edison Company, is with respect to Indian Point Unit 2, subject to the foregoing requirements, and has taken the following steps with respect thereto.

The Company has furnished to the Commission proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association policy (Nuclear Energy Liability Policy, facility form) Nos. NF-100.

Further, the Company executed Indemnity Agreement No. B-19 with the Commission as of January 12, 1962, which was amended to cover its pertinent preoperational fuel storage under license SNM-1108 on March 4, 1969. The Company has paid the annual indemnity fee applicable to preoperational fuel storage.

17.2 <u>Operating License</u>

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from, preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for reactors which have a rated capacity of 100,000 electrical kilowatts or more is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently $82 million.

Accordingly, no license authorizing operation of Indian Point Unit 2 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended and that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operation. The amount of financial protection required for a reactor having the rated capacity of this facility would be $82 million. Consolidated Edison Company will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of $30 per each thousand kilowatts of thermal capacity authorized in its operating license.

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licensees, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

18.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that:

1. The application for facility license filed by the Consolidated Edison Company of New York, Inc., dated December 6, 1965, as amended (Amendments Nos. 9 through 25, dated October 15, 1968, October 13, 1969, October 24, 1969, November 21, 1969, December 29, 1969, January 27, 1970, March 2, 1970, March 30, 1970, April 17, 1970, June 3, 1970, July 14, 1970, July 17, 1970, July 28, 1970, July 29, 1970, August 13, 1970, August 28, 1970, and November 12, 1970, respectively) complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and
2. Construction of the Indian Point Nuclear Generating Unit No. 2 (the facility) has proceeded and there is reasonable assurance that it will be completed, in conformity with Provisional Construction Permit No. CPPR-21, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
3. The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

4. There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
5. The applicant is technically and financially qualified to engage in the activities authorized by this operating license, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
6. The applicable provisions of 10 CFR Part 140 have been satisfied; and
7. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

Prior to any public hearing on the matter of the issuance of an operating license to Consolidated Edison for Indian Point Unit No. 2, the Commission's Division of Compliance will prepare and submit a supplement to this Safety Evaluation which will deal with those matters relating to the status of construction completion and conformaty of this construction to the provisional construction permit and the application. Before an operating license will be issued to Consolidated Edison for Indian Point Unit No. 2, assuming such a license is authorized following the public hearing, the facility must be completed in conformity with the provisional construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Division of Compliance prior to license issuance.

APPENDIX A

CHRONOLOGY OF

REGULATORY REVIEW OF THE CONSOLIDATED EDISON COMPANY

INDIAN POINT NUCLEAR GENERATING PLANT UNIT NO. 2

(SUBSEQUENT TO CONSTRUCTION PERMIT NO. CPPR-21

ISSUED ON OCTOBER 14, 1966)

1.	April 17, 1967	Submittal of Amendment No. 6 containing design information on the Emergency Core Cooling System and other areas as requested by the ACRS in their letter to the Chairman AEC, of 8/16/66.
2.	July 18, 1967	Meeting with applicant to discuss revised design of Emergency Core Cooling System and other areas as per Amendment No. 6.
3.	August 2, 1967	Letter to applicant requesting additional information on subjects addressed by the ACRS in their letter of 8/16/66.
4.	October 16, 1967	Submittal of Amendment No. 7 in response to DRL request of August 2, 1967.
5.	October 31, 1967	Submittal of Amendment No. 8, revised pages for Amendment No. 7.
6.	December 28, 1967	ACRS Subcommittee meeting to discuss emergency core cooling system, reactor pit crucible, primary coolant system, other areas.
7.	January 30, 1968	Submittal of "Report on the Containment Building Liner Plate Buckle in the Vicinity of the Fuel Transfer Canal".
8.	February 2, 1968	Meeting with applicant to discuss content of Amendments No. 6, 7, and 8.
9.	February 13, 1968	Meeting with applicant to complete discussion of February 2, 1968.

10.	March 8, 1968	ACRS Full Committee meeting to discuss Emergency Core Cooling System; reactor internals; primary coolant system, design, fabrication, in-service inspection, and leak detection; core design; reactor pit crucible; and containment liner quality control and stress analysis.
11.	October 15, 1968	Consolidated Edison Company filed application for an Operating License for the IP-2 Plant. Amendment 9, Volumes 1, 2, 3, & 4.
12.	March 5, 1969	AEC-DRL requested additional information on medical and emergency plans.
13.	March 12, 1969	AEC-DRL staff met with Con Ed personnel to discuss scheduling of regulatory review of application for operating license.
14.	April 3, 1969	AEC-DRL staff met with Con Ed personnel to discuss structural and seismic design and tornado protection.
15.	April 16, 1969	AEC-DRL staff met with Con Ed to discuss accidental and normal radioactivity release from the IP-2 plant.
16.	April 28, 1969	Con Ed requested extension of completion date for construction of the IP-2 plant.
17.	May 2, 1969	AEC-DRL staff and Nathan M. Newmark, seismic design consultant, met with Con Ed personnel at the IP-2 site to discuss seismic design and review status of construction and site inspection.
18.	May 19, 1968	AEC-DRL staff issued an order extending completion date for construction of the IP-2 plant to June 1, 1970.

19.	August 4, 1969	Request to applicant for additional information on site and environment, reactor coolant system, containment system, engineered safety features, instrumentation and control, electrical systems, waste disposal and radiation protection, conduct of operations, and accident analysis.
20.	August 22, 1969	AEC-DRL staff requests copies of monitoring reports and status of actions on Fish and Wildlife recommendations.
21.	August 23, 1969	ACRS Subcommittee meeting on tornado protection, emergency planning, permanent in-core instrumentation, adequacy of onsite emergency power, and containment isolation.
22.	September 24, 1969	Meeting with applicant to discuss Westinghouse presentation on power distribution detection and control in Indian Point 2.
23.	October 13, 1969	Submittal of Amendment 10 (Supplement #1) responses to AEC regulatory staff's request of March 5, 1969, on medical plans and partial answers to AEC regulatory staff's request for additional information of August 4, 1969.
24.	October 24, 1969	Submittal of Amendment No. 11, replacement pages and responses to AEC regulatory staff's request for additional information of August 4, 1969, on Sections 1, 4, 5, 6, 7, 12, and 14 of the FSAR.
25.	November 13, 1969	Request for additional information on reactor, reactor coolant system, containment system, engineered safety features, auxiliary and emergency systems, initial tests and operations, and accident analysis.
26.	November 21, 1969	Submittal of Amendment No. 12, additional and replacement pages to be inserted into the FFDSAR and further responses to AEC regulatory staff's request for additional information of 8/4/69 on Sections 1, 4, 7, 8 and 11 of the FFDSAR.

27.	December 10, 1969	Meeting with applicant to review electrical drawings including AC power, DC power, Reactor Protection System, and Engineered Safety Features.
28.	December 30, 1969	Meeting with applicant and Westinghouse Electric Corporation to continue detailed review of electrical drawings including Reactor Protection System and Engineered Safety Features.
29.	January 16, 1970	Meeting with applicant to review and discuss electrical drawings including Reactor Protection System and Engineered Safety Features.
30.	January 21, 1970	Meeting with applicant & Westinghouse Electrical Corporation on technical specifications.
31.	January 27, 1970	Submittal of Amendment No. 14, replacement pages for FSAR & further responses to AEC-DRL questions of 8/4/69 & 11/13/69, chapters 1, 4, 6, 11, 12 & 14.
32.	February 17, 1970	Meeting with applicant for presentation of results of Con Ed's Analysis concerning potential damage to Indian Point 2 and IP-3 from a failure of the IP-1 superheater stack.
33.	March 2, 1970	Submittal of Amendment No. 15, responses to AEC regulatory staff's requests for additional information of 8/4 and 11/13, 1969 and Containment Design Report.
34.	March 10, 1970	Request to applicant for additional financial data.
35.	March 13, 1970	Meeting with applicant to discuss questions concerning core heat transfer and burnout limits, fuel element performance and ECCS performance during a LOCA.

36.	March 19, 1970	Meeting with applicant, Westinghouse presentation on iodine removal system for IP-2.
37.	March 26, 1970	Meeting with applicant to discuss analysis of fresh water flood and changes to electrical systems.
38.	March 30, 1970	Submittal of Amendment No. 16, additional and replacement pages for the FSAR and further responses to the AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
39.	April 25, 1970	ACRS Subcommittee meeting and meeting with applicant on instrumentation and control, and anticipated transients with failure to scram.
40.	April 17, 1970	Submittal of Amendment No. 17, additional and replacement pages to be inserted into the FSAR and further responses to AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
41.	April 29, 1970	Meeting with applicant to discuss seismic and structural design questions for IP-2.
42.	May 5, 1970	Meeting with applicant to discuss failure mode analysis of the engineered safety feature manual actuation panel.
43.	May 11, 1970	ACRS Subcommittee meeting at the Indian Point 2 site to discuss instrumentation and control and Electrical Systems.
44.	May 12, 1970	AEC issued Order extending completion date for construction of the IP-2 plant to June 1, 1971.
45.	May 28, 1970	ACRS Subcommittee meeting to discuss loss-of-coolant accident, anticipated transients with failure to scram.
46.	June 3, 1970	Submittal of Amendment No. 18, additional and revised pages for the FSAR in response to AEC regulatory staff request for additional information.

47.	June 11, 1970	ACRS full Committee meeting to consider design of engineered safety feature manual actuation panel and operation with less than four loops.
48.	June 17, 1970	Meeting with applicant to discuss consequences of turbine missiles, sensitized stainless steel control room accident dose, hydrogen recombiner.
49.	July 15, 1970	Submittal of Amendment No. 19 (Supplement 10), additional and revised pages for the FSAR and Flooding Evaluation report.
50.	July 20, 1970	Submittal of Amendment No. 20, (Supplement 11) proposed Technical Specifications.
51.	July 24, 1970	Request for additional information on emergency core cooling, reactor coolant system, instrumentation and control, electrical systems, conduct of operations and accident analysis.
52.	July 28, 1970	Submittal of Amendment No. 21, Con Ed Annual Report.
53.	July 28 and 29, 1970	ACRS Subcommittee meeting to discuss technical specifications, flood protection, Unit No. 1 superheater stack failure and containment sprays.
54.	July 30, 1970	Submittal of Amendment No. 22, (Supplement 12), revised pages for FSAR in response to request for additional information.
55.	August 7, 1970	Meeting with applicant to discuss technical specifications.
56.	August 13, 1970	ACRS full Committee meeting to discuss the matters addressed in our July 2, 1970 report.
57.	August 14, 1970	Submittal of Amendment No. 23 (Supplement 13), answers to request for additional information issued July 24.

58.	August 18, 1970	Meeting to discuss licensed operator requirements.
59.	August 28, 1970	Submittal of Amendment No. 24 (Supplement 14). Revised pages to the FSAR.
60.	September 1, 1970	Meeting with applicant regarding performance of Emergency Core Cooling System.
61	September 9, 1970	Meeting with the applicant to discuss Technical Specifications.
62.	October 21, 1970	Request to applicant for a report on analysis of laminations in base plate material of the IP-2 pressurizer.
63.	October 29, 1970	Meeting with applicant to review technical specifications for the Indian Point 2 plant.
64.	November 1970	Submittal of Amendment 25 (Supplement 15), changes to technical specifications and to FSAR.

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

SEP 23 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by Consolidated Edison Company of New York, Inc., for authorization to operate the Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the Committee's 95th, 98th, 122nd, and 124th meetings, and at Subcommittee meetings on August 23, 1969, March 13, 1970, April 25, 1970, May 28, 1970, July 28-29, 1970, and September 15, 1970. Subcommittees also met at the site on December 28, 1967 and May 11, 1970. The Committee last reported on this project to you on August 16, 1966. During the review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants, and with representatives of the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Indian Point site is located in Westchester County, New York, approximately 24 miles north of the New York City limits. The minimum radius of the exclusion area for Unit No. 2 is 520 meters and Peekskill, the nearest population center, is approximately one-half mile from the unit. Also at this site are Indian Point Unit 1, which is licensed for operation at 615 MWt, and Unit 3, which is under construction.

The applicant has re-evaluated flooding that could occur at the site in the event of the probable maximum hurricane and flood, in the light of more recent information, and has concluded that adequate protection exists for vital components and services.

Additional seismic reinforcement being provided for the Indian Point Unit No. 1 superheater building and removal of the top 80 ft. of the superheater stack will enable the stack to withstand winds in the range of 300-360 mph corresponding to current tornado design criteria. Since the reinforcement of the superheater building, which supports the stack, enables the stack to resist wind loads of a magnitude most likely to be experienced from a tornado, the Committee believes that removal of the top 80 ft. of the stack, to enable it to resist the maximum effects from a tornado, may be deferred until a convenient time during the next few years, but prior to the commencement of operation of Indian Point Unit No. 3. The applicant has stated that truncation of the stack will have no significant adverse effect on the environment.

The Indian Point Unit No. 2 is the first of the large, four-loop Westinghouse pressurized water reactors to go into operation, and the proposed power level of 2758 MWt will be the largest of any power reactor licensed to date. The nuclear design of Indian Point Unit No. 2 is similar to that of H. B. Robinson with the exception that the initial fuel rods to be used in Indian Point Unit No. 2 will not be prepressurized. Part-length control rods will be used to shape the axial power distribution and to suppress axial xenon oscillations. The reactor is designed to have a zero or negative moderator coefficient of reactivity, and the applicant plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses related to these tests.

Unit 2 has a reinforced concrete containment with an internal steel liner which is provided with facilities for continuous pressurization of weld and penetration areas for leak detection, and a seal-water system to back up piping isolation valves. In the unlikely event of an accident, cooling of the containment is provided by both a containment spray system and an air-recirculation system with fan coolers. Sodium hydroxide additive is used in the containment spray system to remove elemental iodine from the post-accident containment atmosphere. An impregnated charcoal filter is provided to remove organic iodine.

Major changes have been made in the design of the emergency core cooling system as originally proposed at the time of the construction permit review. Four accumulators are provided to accomplish rapid reflooding of the core in the unlikely event of a large pipe break, and redundant pumps are included to maintain long-term core cooling. The applicant has analyzed the efficacy of the emergency core cooling system and concludes that the system will keep the core intact and the peak clad temperature well below the point where zircaloy-water reaction might have an adverse effect on clad ductility and, hence, on the continued structural integrity of the fuel elements. The Committee believes that there is reasonable assurance that the Indian Point Unit No. 2 emergency core cooling system will perform adequately at the proposed power level.

The Committee concurs with the applicant that the reactor pit crucible, proposed at the time of the construction permit review, is not essential as a safety feature for Indian Point Unit No. 2 and need not be included.

To control the concentration of hydrogen which could build up in the containment following a postulated loss-of-coolant accident, the applicant has provided redundant flame recombiner units within the containment, built to engineered safety feature standards. Provisions are also included for adequate mixing of the atmosphere and for sampling purposes. The capability exists also to attach additional equipment so as to permit controlled purging of the containment atmosphere with iodine filtration. The Committee believes that such equipment should be designed and provided in a manner satisfactory to the Regulatory Staff during the first two years of operation at power.

The applicant plans to install a charcoal filter system in the refueling building to reduce the potential release of radioactivity in the event of damage to an irradiated fuel assembly during fuel handling. This installation will be completed by the end of the first year of full power operation.

The reactor instrumentation includes out-of-core detectors, fuel assembly exit thermocouples, and movable in-core flux monitors. Power distribution measurements will also ordinarily be available from fixed in-core detectors.

The applicant has proposed that a limited number of manual resets of trip points, made deliberately in accordance with explicit procedures, by approved personnel, independently monitored, and with settings to be calibrated and tested, should provide an acceptable basis for the occasional operation of Indian Point Unit No. 2 with only three of the four reactor loops in service. The Committee concurs in this position.

The applicant stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefulness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

The reactor includes a delayed neutron monitor in one hot leg of the reactor coolant system to detect fuel element failure. Suitable operability requirements will be maintained on the several sensitive means of primary system leak detection.

A conservative method of defining pressure vessel fracture toughness should be employed that is satisfactory to the Regulatory Staff.

The applicant stated that existing experimental results and analyses provide considerable assurance that high burnup fuel of the design employed will be able to undergo anticipated transients and power perturbations without a loss of clad integrity. He also described additional experiments and analyses to be performed in the reasonably near future which should provide further assurance in this regard.

The Committee has, in recent reports on other reactors, discussed the need for studies on further means of preventing common failure modes from negating scram action, and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has provided the results of analyses which he believes indicate that the consequences of such transients are tolerable with the existing Indian Point Unit No. 2 design at the proposed power level. Although further study is required of this general question, the Committee believes it acceptable for the Indian Point Unit No. 2 reactor to operate at the proposed power level while final resolution of this matter is made on a reasonable time scale in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept advised.

Other matters relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS letters should, as in the case of other reactors recently reviewed, be dealt with appropriately by the Staff and the applicant in the Indian Point Unit No. 2 as suitable approaches are developed.

The ACRS believes that, if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit No. 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Original Signed by
Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References attached.

References - Indian Point Nuclear Generating Unit No. 2

1. Amendment No. 9 to Application of Consolidated Edison Company of New York for Indian Point Nuclear Generating Unit No. 2, consisting of Volumes I - IV, Final Safety Analysis Report, received October 16, 1968
2. Amendments 10 - 20 to the License Application
3. Amendments 22 - 24 to the License Application

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis Report
Volumes I, II, III and IV dated October 15, 1968

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
November 29, 1968

As pointed out in our comments of October 29, 1965 on Unit No. 2, a primary influence on the meteorological statistics of the Indian Point site seems to be its location in a river valley about a mile wide with terrain rising 600 to 1000 feet on either side. Consequently, wind directions follow a pronounced diurnal cycle with daytime, unstable (lapse) flow in the upriver direction and nighttime, stable flow in the downriver directions. The report documents a 42.4 percent inversion frequency, but it should also be pointed out that inversion conditions are largely confined to the nighttime, downriver flow lasting about 12 hours before changing to lapse or upriver flow. Figure 2.6-1, although in terms of average vectors, shows the marked wind reversals at sunset and sunrise and the rather persistent, channeled flow that can occur during the middle of the night (see the mean direction between 0200 and 0800 hours). The mean wind speeds during this persistent period is about 2.5 m/sec which indicates that 50 percent of the time inversion wind speeds could be less than 2.5 m/sec.

In the absence of specific, joint-frequency wind speed and direction persistence data from the site, a reasonably conservative meteorological model would be to assume for a ground release a 1 m/sec wind speed under inversion conditions in a persistent downriver direction for a period of 8 hours. Taking into account the likelihood of a diurnal wind reversal, a very conservative assumption would be to allow the plume centerline to meander over a 22-1/2° arc under the same conditions for the remainder of the 24-hour period. Again, with no specific on-site wind persistence data, the conservative assumption has been made.

The amount of additional atmospheric diffusion because of the building turbulence can be assessed by the virtual point source expression $[(x + x_o)/x]^{1.5}$ as used by the applicant, which for a value of x_o = 430 m amounts to a factor of 2.5 at the site boundary (520 m) and 1.6 at the low population boundary (1100 m). These values are in close agreement with the method of using a shape factor of 1/2 and a building cross-section of 2000 m^2.

In summary, from data presently available, it would seem reasonably conservative to assume a persistent wind direction for an 8-hour period under inversion conditions and a 1 m/sec wind speed. With the added assumption of a building wake shape factor of 1/2 and a cross-sectional area of 2000 m^2, the resulting 0-8 hr relative concentration would be 6.6×10^{-4} sec m3 at the site boundary and 3.7×10^{-4} at the low population boundary. From Table 14.3.5-3 one can calculate that the applicant's model for the 0-8 hr period results in an average relative concentration of 4.8×10^{-4} and 2.4 sec m^{-3} at the site and low population boundary, respectively.

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis
Amendment No. 12 dated November 21, 1969, and
Amendment No. 14 dated January 27, 1970

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
February 17, 1970

The original documentation of the Indian Point site during the period 1955-1957 indicates that at the 100-ft. height the annual prevailing wind direction is from the north northeast and that in the sector from 22.5 to 42.5 degrees the frequency of inversion, neutral and lapse conditions was 6, 2, and 1 percent, respectively. Within this sector, the shortest site boundary is approximately in a direct line through Units 2 and 3 at a distance of 610 and 380 m, respectively, as measured from figure 2.2-2. It is about 500 m from the Unit 1 stack to this common boundary point. The nearest site boundary, regardless of sector, is where the property line intersects the downriver edge of the site. Although this point is at a distance of 580 m from Unit 2, it is not in the most prevalent wind direction by a considerable amount.

To compute the average annual dilution factor we have assumed the frequencies listed above, averaged over a 20-degree sector with a wind speed of 2, 4 and 5 m/sec, respectively, for inversion (Type F), neutral (Type D), and lapse (Type B) conditions. Assuming no building wake effect our results show the applicant's values for Units 1 and 2 to be reasonably conservative. In the case of Unit 3 we compute an average annual dilution factor of 2.9×10^{-5} sec m^{-3} as compared to the applicant's value of 1.6×10^{-5} sec m^{-3}. The only explanation we have for the ESSA value being twice as high is the use of the building wake effect in the applicant's assumptions.

It is our view that the use of the building wake effect in the long-term average diffusion equation, as was done by the applicant, is inappropriate. It does not seem logical that for the same atmospheric conditions the Sutton equation on page Q 11.10-1 for the long-term model gives <u>more</u> credit for building wake effect than the equivalent short-term model on p. Q 11.10-2. For example at $x = 400$ m assuming $x_o = 400$ m and $n = 0.5$, the building wake effect, $[(x+x_o)/x_o]^{2-n/2}$, for the long-term equation is 3.4 whereas for the effect in the short-term equation, $[(x+x_o)/x_o]^{2-n}$, the value is 2.8. It is the larger exponent in the former that makes the difference. Also, the fact that one averages in the horizontal dimension over a sector essentially would nullify any added dilution in that dimension because of wake effect.

APPENDIX D

DEPARTMENT OF THE ARMY
COASTAL ENGINEERING RESEARCH CENTER
5201 LITTLE FALLS ROAD, N.W.
WASHINGTON, D.C. 20016

CERER

21 November 1969

Mr. Roger S. Boyd
Asst. Director for Reactor Projects
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Boyd:

Reference is made to your letters regarding Docket Nos. 50-247, 50-286, 50-342, and 50-343, Consolidated Edison Company of New York's proposed Indian Point Nuclear Generating Units No. 2 and No. 3, and Units No. 4 and No. 5 which are contiguous to Indian Point plant site.

Pursuant with our arrangements, Mr. R. A. Jachowski and Mr. B. R. Bodine of CERC have reviewed all pertinent information contained in the reports from the standpoint of establishment of a design water level. This included the review of the storm surge associated with the Probable Maximum Hurricane (PMH) and wind wave analysis.

We concur with the applicant's finding that the design water level should be 14.5 feet above the mean sea level datum for Units, Nos. 2, 3, 4 and 5. Although this value is acceptable, there are compensating errors in routing procedure employed.

If you have any further questions regarding this matter please let us know.

Sincerely yours,

Edward M. Willis

EDWARD M. WILLIS
Lieutenant Colonel, CE
Director

APPENDIX E

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

SEP 16 1970

Mr. Harold Price
Director of Regulation
U.S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by R. C. DeYoung is a review of the flood information presented in Amendment No. 19 to the Final Safety Analysis Report for Unit No. 2 Indian Point Nuclear Generating Station. It is presumed that the flood levels for all 3 units at the Indian Point Stations will be based on this amendment. Copies of our earlier reviews, for Unit No. 2 (Aug. 15, 1966) prepared by E. L. Meyer, and for Unit No. 3 (January 6, 1969) prepared by P. J. Carpenter, are attached.

This review was prepared by P. J. Carpenter and has been discussed with members of your staff. We have no objection to your making this review a part of the public record.

Sincerely yours,

W. A. Radlinski

Acting Director

Enclosures

Consolidated Edison Company of New York Inc.
Indian Point Nuclear Generating Station Unit No. 2
Docket No. 50-247

The probable maximum flood as defined by the U.S. Army Corps of Engineers, at the site, has been calculated as 1,[illegible],000 cubic feet per second. This discharge is approximately five times greater than the maximum observed flood at Green Island, and is approximately twice the maximum discharge observed for nearby like-sized drainage basins which appear to exhibit similar runoff characteristics. The stage for the maximum probable flood at the site, computed using standard step-backwater procedures, is given as varying between 13.4 and 14.0 ft msl (mean sea level) depending on concurrent tide levels at the Battery. It is shown that none of the dams on the Hudson River and its tributaries would fail during the probable maximum flood. The above results were obtained using conservative assumptions and appear to be reasonable.

The analyses show that the occurrence of the probable maximum flood on Esopus Creek would cause failure of Ashokan Dam some 75 miles upstream of the site. To establish a flood design level at Indian Point various combinations of the following factors were considered: 1) the flow resulting from the Ashokan Dam failure, 2) various concurrent Hudson River flood flows, and 3) various concurrent tide levels at the Battery. The results of these combinations of factors were compared with the stage of the probable maximum flood (14.0 ft msl) and the stage resulting from the probable maximum hurricane plus spring high tide (14.5 ft msl). The most critical combination investigated consisted of the flows from the Ashokan Dam failure caused by the probable maximum flood on Esopus Creek, the concurrent standard project flow (one half the probable maximum flood), the concurrent stage at the Battery corresponding to the standard project hurricane tide level and wind waves of one foot at the site. This stage is given as 15.0 ft msl. The lowest floor elevation of Unit No. 2 is given as 15.25 ft msl.

Other combinations of the above-mentioned factors, such as Ashokan Dam failure and the standard project hurricane or floods larger than the standard project flood on the Hudson River, could produce higher stages at the site. Depending on the degree of conservatism desired, any of these higher stages could also be selected as the design flood level. However, the stage for the combination selected for the design flood level exceeds those given for the probable maximum flood or probable maximum hurricane when these are considered as independent events.

APPENDIX F

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Consolidated Edison Company of New York, Inc.

Docket No. 50-247

by

N. M. Newmark
and
W. J Hall

Urbana, Illinois

20 August 1970

1

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

INTRODUCTION

This report is concerned with the structural adequacy of the containment structures, piping, equipment and other critical components for the Indian Point Nuclear Generating Unit No. 2 for which application for a construction permit and an operating license has been made to the United States Atomic Energy Commission by the Consolidated Edison Company of New York, Inc. The facility is located on the east bank of the Hudson River at Indian Point, village of Buchanan, in upper Westchester County, New York. The site is about 24 miles N of the New York City boundary and 2.5 miles SW of Peeksill, New York.

This report is based on a review of the Final Facility Description and Safety Analysis Report (Ref. 1) and the containment design report (Ref. 2). The report also is based in part on the discussion and inspection resulting from the visit to the site on 2 May 1969 by N. M. Newmark and W. J. Hall in conjunction with Mr. K. Kniel and Mr. M. McCoy of AEC-DRL. A number of topics were discussed with the applicant and his consultants at the time of this visit, and subsequently additional information has become available through supplements to the FSAR and through discussions with the personnel of DRS, DRL, and the applicant and his consultants. A discussion of the adequacy of the structural criteria presented in the Preliminary Safety Analysis Report is contained in our report of August 1966 (Ref. 3), and unless otherwise noted no comment will be made in this report concerning points covered there.

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The design criteria for the containment system and Class I components for this plant called for a design to withstand a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration coupled with other appropriate loadings to provide for containment and safe shut down. The plant was also to be designed for an Operating Basis Earthquake of 0.1g maximum horizontal ground acceleration simultaneously with the other appropriate loads forming the basis of containment design.

COMMENTS ON ADEQUACY OF DESIGN

Dynamic Analyses

(a) Containment Building. The answer to Question 1.9 of the FSAR indicates that only the containment building, the primary auxiliary building, and the electric cable tunnel were designed with the use of semi-formal dynamic analyses. A description of the method of analysis employed is given briefly in Section 5.1.3.8 of the FSAR and in Section 3.1.5 of the containment design report. The procedure employed involved a calculation of the fundamental frequency and mode shape by use of a modified Rayleigh method. The base shear for the structure was computed from the period and the spectral response corresponding to the appropriate degree of damping. The base shear was then applied as a loading to the structure as an inverted triangular loading. The shears at the nodes were used to calculate the moments and displacements at various points in the structure. For the structures involved it is believed that the approach leads to a design which is reasonably adequate.

A similar approach was followed for the primary auxiliary building as described in the answer to Question 1.9. It is noted there that a one-third increase over working stress was allowed in the design of the bracing in the case of the Design Basis Earthquake. This stress is below yield, and it is believed that the design will prove to be satisfactory.

(b) Other Buildings and Equipment. The discussion presented in answer to Question 1.9 of the FSAR for other buildings and equipment such as the control building, fan house, intake structure, etc., indicate that a refined static approach was used, which involves employing the peak value from the appropriate response spectrum curve for a given value of damping and multiplying this by the appropriate mass to obtain the inertial loading. From the description given for the various buildings and items of equipment, and the modeling techniques employed, it is concluded that the inertial loadings used in design are reasonably close to those that might be obtained with a more sophisticated analysis and lead to reasonable design values.

The submission in Question 1.3 of Supplement 13 indicates that the Turbine Building, and Fuel Storage Building Structure above the Fuel Storage Pit were reanalyzed by a multi-degree-of-freedom modal dynamic analysis method to check their adequacy. As a result of this reanalysis, the applicant advises that certain structural modifications will be made to columns and cross bracing in the Turbine Building to insure that it can withstand the DBE. The superstructure of the fuel storage building was ascertained to be adequately designed, without modification to withstand the effects of the DBE. The applicant states that reanalysis of the strengthened turbine building and superheater building for Indian Point No. 1 does not significantly affect the responses calculated for the original structures.

(c) Piping Analysis. The method used by the applicant for analysis of the piping, as described in the answer to Question 1.6 of the FSAR, is the same as was used in Ginna. The peak ground response spectrum value for 0.5 percent damping was used, applied as static accelerations in each direction

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separately, and the resulting stresses superposed. It was assumed by the applicant that the piping was supported along rigid systems and therefore not subjected to amplified ground motion at points of support. The system was analyzed with the anchors and supports as actually used, according to the discussion presented to us during the time of our visit in May, 1969. It was the view of the applicant that the thermal motions were greater than any differential ground displacements and the latter therefore are not critical items in the design. In answer to Question 1.13 (Suppl. 13) the applicant advises that relative seismic displacement was considered for the main steam lines, where the largest relative displacements are expected; stress differentials of less than 10% resulted. Also, seismic supports installed to date are those specified in the design and employed in the analyses; where deviations in supports must occur, reanalysis will be carried out. These results and approaches appear satisfactory to us.

Since this plant was designed before recent developments and changes in piping design specifications, the 1968 ASME Addenda were not applied. Blow-down and earthquake were considered as separate items and not combined in this design. We are advised that the response to Question 1.9 of Supplement 12 states that a review of the Indian Point 3 reactor coolant system which is identical to Indian Point 2, for combined earthquake and blow-down indicates that the design is adequate.

It is stated in the answer to Question 1.6 of the FSAR that the approach resulted in a seismic design load approximately equal to 0.60W horizontally and 0.40W vertically taken simultaneously. It is further stated that for the Design Basis Earthquake the sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code

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allowable stresses. In a similar manner the stresses in the pipe supports and hangers were limited to 1.2 times code allowable stresses.

The applicant originally made use of the maximum spectrum value only and no modal analyses were made; in other words only a static analysis with uniform accelerations was made. Consideration was not given to modified distribution of the inertial loading to take account of the combination of modal effects.

The response to Question 1.9 of Supplement 8, describing more detailed analyses of the reactor coolant system, feedwater lines, surge lines and typical steam lines by more formal methods as carried out later lends confirmation to the adequacy of the design. On this basis, there is reason to believe that the design is adequate.

Backfill Surrounding Containment Vessel

Nine feet of crushed rock backfill was placed between the external wall of the reinforced concrete containment vessel and the retaining wall holding back the rock on the uphill side. This crushed rock backfill is drained at the bottom to avoid water pressure against the containment structure. The fill is approximately 60 to 70 feet higher on one side of the structure than on the other because of the slope of the rock surface. The design, as discussed in Section 3.1.5 of the containment design report, considered local inertial forces of loose rock as an added loading against the containment pressure vessel, and also considered passive pressures caused by failure of the rock along the surface behind the retaining wall. The localized loadings from these forces were considered in the design of the containment structure and the discussion presented in the containment design report provides reasonable assurance that the containment vessel is capable of resisting these localized forces.

Class I Equipment in Structures other than Class I

The turbine building is Class III and not designed for earthquake loadings. The answer to Question 1.3 of the FSAR indicates that the only Class I structures and components which are so located that they could be endangered by failure of Class III structures are the control building, main steam piping and feedwater piping, all of which could possibly be endangered by the Class III turbine building. It is further indicated there that no special provisions have been provided for protection except in the case of the main steam and feedwater lines up to the isolation valves, which are protected by the shield wall and the structural frame at the north end of the shield wall. Since these are located near the braced end of the turbine building, it is not anticipated by the applicant that there will be any structural failure in this area. Our judgment as to the adequacy of this aspect of the design is based on the statement given in the application. And, in this respect, the answer to Question 1.3 (Supplement 13) which describes the analysis and strengthening of the Turbine Building and Superheater Building for Indian Point Unit No. 1, and their ability to withstand the DBE, should give additional protection for the control room.

It is further stated that the only Class III crane whose failure could endanger any Class I function is the fuel storage building crane and that the failure of this crane will not impair a safe and orderly shutdown. The answer to Question 1.3 (Suppl. 13) indicates that the only potential for crane lift off will be in the unloaded condition with the trolley parked over the support; the applicant advises that the unloaded crane will not be parked over the pool, so no hazard exists. It is also noted in the answer to Question 1.1.3 that the manipulator crane in the containment building, a Class III crane, is restrained from overturning and will not endanger Class I structures.

Deformation Criteria

The general stress criteria applicable to the seismic design are summarized in Appendix A of the FSAR. The statement given on page, A3 of Appendix A states that for all components, systems and structures classified as Class I, the primary steady state stresses, when combined with seismic stresses resulting from the response to the Design Basis Earthquake, are limited so that the function of the component system or structure shall not be impaired so as to prevent a safe and orderly shut-down of the plant.

We were advised at the time of our inspection of the plant in May 1969 that, for normal loadings plus the Operating Basis Earthquake, the intention was to use code allowables plus the 20 percent increase for transient conditions on Class I components and systems. For the Design Basis Earthquake and blow-down, basically the same criteria were used, although originally it had been planned to adopt higher allowables going into the plastic range using the code for faulted conditions. In actuality, as described in the answer to Question 1.7 of the FSAR, the allowable stresses in the case of the Design Basis Earthquake were limited to the yield point, or slightly below (see answer to Question 1.3 of Supplement 13).

The only references that we note where there was a calculation of stresses exceeding the yield point were at several places in the containment design report where it was mentioned that the calculations indicate that there could be possible local yielding of the liner under certain loading combinations, but that this would be limited and not be expected to be of a nature as to cause concern with regard to the integrity of the liner.

Reactor Internals

The mechanical design and evaluation of the reactor core and internals is described generally in Section 3.2.3 of the FSAR. From the discussion given it appears that the core support structure and core barrel have been designed with proper attention to support points and limitations of motions. The design criteria for the internals themselves, and specifically with reference to deflections under abnormal operation, are given in Table A.3-2 of the FSAR. These appear reasonable and should provide an adequate margin of safety.

Large Penetrations

A finite element analysis of the large penetrations in the containment vessel was made by the Franklin Institute and a description of the analysis and the results obtained is presented in the containment design report. Several analyses were made for different load combinations, and in addition a number of hand calculations were made to check the order of magnitude of the expected forces and stresses and to verify that the results were reasonable Our review of the material presented, to the extent possible, indicates that the penetration design is adequate.

Splices in Large Reinforcing of Bars

Cadweld splices were used in general in the construction of the containment vessel. We were advised that the early splices, about 10 percent of the total, were made with a bronze base, and the remaining 90 percent were made with ferritic base filler metal. Around the hatch opening, we observed that there was approximately a three foot stagger of adjacent splices, but in questioning we learned that there may not be such a stagger over other areas of the containment vessel. Lack of stagger of adjacent splices could the siding of the control room can resist wind velocities up to 162 mph, and the girts (supporting the panels) will fail at 0.62 psi negative pressure; the building is protected by other buildings on the south and west.

Steel Liner and Containment Vessel

The analyses that have been carried out with regard to the liner are summarized in the FSAR and some additional information is presented in the containment design report. It is our understanding that where bulges of the liners occurred during construction, of less than 2 in., nothing was done to correct the bulges. However, when bulges were 2 in. or greater the liner was pushed back into a position of not more than 2 in. away from its intended position, and additional studs were used to anchor the liner in place. Temporary bracing was employed to hold it in position until the concrete was cast. Because of the foregoing, and since the temperature rise in the lower part of the structure in the liner is reduced by the use of insulating material, it is not expected that the departures from the intended original surface will lead to any difficulties.

Proof Test Procedures and Instrumentation

It is our understanding that a detailed description of the proof test procedures is to be submitted at a later date. At the time of our visit in May 1969 it was proposed by the applicant that strain readings be taken only on the liner around the penetrations. We suggested that additional readings be made which would include diameter changes of the penetrations and other measurements that can be made conveniently and without excessive expense to provide evidence that the design meets the design criteria. Fig. 5.13-4 suggests that such readings will be made. In any event, an interpretative report on the measurements that are taken should be provided and should be correlated with the calculations to provide evidence of validity of the design calculations.

Protection of Pipe Lines for Service Water

We were advised that pipelines for service water are embedded in the ground without any special protection. However, there appear to be alternate lines, although they are generally in the same location and/or trenches. In view of the foundation conditions surrounding the plant, and since there is no indication of previous fault motion or potential faulting, this design approach appears to be adequate. If redundancy in critical water supply is desired, it would be preferable to have separate water lines following independent routes.

Seismograph Installation

The answer to Question 1-1 of Supplement 3 indicates that one seismograph will be installed in the yard area, to provide further evidence of the extent of seismic excitation to which the plant might be subjected if an earthquake occurs. This is acceptable to us.

Containment Design Report

The containment design report, prepared for the applicant by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, has proven to be helpful in arriving at an evaluation of many of the factors inherent in the design. The tables presented are useful in helping to arrive at decisions as to the adequacy of the design; we commend those responsible for the preparation of this summary type material.

We should like to encourage this type of approach to studies of the containment, structures, piping, equipment and other Class I items. We should like to urge that attention be given also to summaries and tabulation of the most important information, in terms of stresses and deformations, including the sources of the various stress components, how they were combined, and related discussion and explanatory material (including figures) which would lend itself to a much better basis for judgment as to the adequacy of design of nuclear facilities in general.

CONCLUDING REMARKS

On the basis of the information made available to us concerning the Class I structures, piping, reactor internals, and other Class I items, it is our belief that the plant possesses a reasonable margin of safety to meet the original design requirements, including the imposed Design Basis Earthquake loading conditions.

REFERENCES

1. "Final Facility Description and Safety Analysis Report -- Vols. I through V including Supplements 1, 2, 4, 5, 6, 7, 8 and 13," Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., AEC Docket No. 50-247, 1969 and 1970.

2. "Containment Design Report," for Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., prepared by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, March 1969. (Labeled Final Draft)

3. "Adequacy of the Structural Criteria for Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2," by N. M. Newmark and W. J. Hall, August 1966.

W. J. Hall

APPENDIX G

UNITED STATES
DEPARTMENT OF THE INTERIOR
OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

OCT 16 1970

Dear Mr. Chairman:

Pursuant to Section 5 of Public Law 89-605 as amended and other authorizations, we are presenting the views of the Department of the Interior in the matter of the application by the Consolidated Edison Company for an operating license for Indian Point Nuclear Generating Unit No. 2, Buchanan, New York, AEC Docket No. 50-247 (Amendment No. 9). The following comments incorporate those submitted by the Federal Water Quality Administration, the Fish and Wildlife Service and the Bureau of Outdoor Recreation.

The unit under review is the second of three units completed or being constructed at the Indian Point site. We note that applications for construction permits for two more units to be located approximately one mile south of the Indian Point site were made in June 1969.

The Department of the Interior does not object to the issuance of the operating license to the Consolidated Edison Company for Unit No. 2 of the Indian Point Nuclear Power Plant. Our position is based upon the firm commitment by the Company as expressed in its responses to the Atomic Energy Commission that it will meet the water quality standards applicable to the receiving waters and that it will take whatever steps are necessary to mitigate any harmful effects that operation of the plant may have on the fishery resources of the Hudson River and tributary waters.

The Company should be commended for the cooperation it has extended to representatives of this Department during the course of our review. The studies which the Consolidated Edison Company is presently engaged in indicate the Company's concern for the potential damages to the environment that could result from operation of this unit and the others planned at and in the vicinity of Indian Point.

We are pleased to note that the Company has made provisions to open part of its land holdings for compatible public recreation use. We express the hope that the Company's public use plans will be finalized and fully implemented at the earliest possible time.

Consolidated Edison has initiated or participated in a number of studies to determine the effects of both radiological and thermal discharges from the Indian Point reactors upon both the temperature distribution and the aquatic life of the Hudson River through its consultants, Quirk, Lawler and Matusky Engineers, and the Alden Research Laboratories of Worcester Polytechnic Institute. The Company has conducted mathematical studies of the probable temperature in the River and has checked these estimates with hydraulic, model studies and actual field studies. In addition, Consolidated Edison has supported several independent but coordinated studies of the micro-organisms and aquatic life in the Hudson River and the probable effects of temperature and salinity changes upon them in the vicinity of the Indian Point Plant.

These studies are continuing and have been and will be helpful in assessing the effects of the Indian Point Unit No. 2 and of the other thermal plants which are proposed for construction on the shores of the Hudson River in the vicinity of Indian Point.

We have been provided information on plans for environmental monitoring of radiological and thermal releases proposed as a part of the operating license application. We understand that the plans for water quality monitoring, including radiological concentrations in the environment in microscopic and macroscopic aquatic life are acceptable to the State of New York. They appear reasonable and are considered generally acceptable to the Department of the Interior.

Through the monitoring programs the Company should have the necessary information to control its activities in a manner that will not violate applicable New York State as well as Federal water quality standards, recommendations of any enforcement conference or hearing board approved by the Secretary or order of any court under Section 10 of the Federal Water Pollution Control Act, and/or other State and Federal water pollution control regulations.

In view of the extensive and valuable fish and wildlife resources in the project area, it is imperative that every possible effort be made to safeguard these resources. Therefore, it is recommended that the Consolidated Edison Company be required to:

1. Continue to work closely with the Department of the Interior, New York State Department of Health, and other interested State and Federal agencies in developing plans for radiological surveys.

2. Conduct pre-operational radiological surveys as planned. These surveys should include but not be limited to the following:

   a. Gamma radioactivity analysis of water and sediment samples collected within 500 feet of the reactor effluent outfall.

   b. Beta and Gamma radioactivity analysis of selected plants and animals (including mollusks and crustaceans) collected as near the reactor effluent outfall as possible.

3. Prepare a report of the pre-operational radiological surveys and provide five copies to the Secretary of the Interior prior to project operation.

4. Conduct post-operational radiological surveys similar to that specified in recommendation (2) above, analyze the data, and prepare and submit reports every six months during reactor operation or until it has been conclusively demonstrated that no significant adverse conditions exist. Submit five copies of these reports to the Secretary of the Interior for distribution to appropriate State and Federal agencies for evaluation.

In addition to the above, the Atomic Energy Commission should urge the Consolidated Edison Company to:

1. Meet with the Department of the Interior, New York State Department of Environmental Conservation, New York State Department of Health, and other interested Federal and State agencies at frequent intervals to discuss new plans and evaluate results of the Company's ecological and engineering studies;

2. Conduct post-operational ecological surveys planned in cooperation with the above named agencies, analyze the data, prepare reports, and provide five copies of these reports to the Secretary of the Interior every six months or until the results indicate that no significant adverse conditions exist;

3. Construct, operate, and maintain fish protection facilities at the cooling water intake structure as needed to prevent significant losses of fish and other aquatic organisms; and

4. Modify project structures and operations including the addition of facilities for cooling discharge waters and reducing concentrations of harmful chemicals and other substances as may be determined necessary.

We appreciate the opportunity to provide these comments.

Sincerely yours,

Walter J. Hickel

Secretary of the Interior

Honorable Glenn T. Seaborg
Chairman, United States
Atomic Energy Commission
Washington, D. C. 20545

APPENDIX H
CONSOLIDATED EDISON COMPANY OF NEW YORK
DOCKET NO. 50-247
FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended Dec. 31		
	1969	1968	1965
Long-term debt	$1,981.6	$1,901.6	$1,711.0
Utility plant (net)	3,793.3	3,583.6	3,169.5
Ratio - debt to fixed plant	.52	.53	.54
Utility plant (net)	3,793.3	3,583.6	3,169.5
Capitalization	3,818.4	3,667.6	3,228.1
Ratio - net plant to capitalization	.99	.98	.98
Stockholders' equity	1,836.7	1,766.0	1,517.1
Total assets	4,069.6	3,845.4	3,387.0
Proprietary ratio	.45	.46	.45
Earnings available to common equity	93.1	95.7	89.9
Common equity	1,210.2	1,139.0	1,072.1
Rate of return on common equity	7.7%	8.4%	8.4%
Net income	127.2	128.5	111.8
Stockholders' equity	1,836.7	1,766.0	1,517.1
Rate of return on stockholders' equity	6.9%	7.3%	7.4%
Net income before interest	198.0	193.9	168.4
Liabilities and capital	4,069.6	3,845.4	3,387.0
Rate of return on total investment	4.9%	5.0%	5.0%
Net income before interest	198.0	193.9	168.4
Interest on long-term debt	84.3	77.0	62.7
No. of times fixed charges earned	2.3	2.5	2.7
Net income	127.2	128.5	111.8
Total revenue	1,028.3	982.3	840.2
Net income ratio	.124	.131	.133
Operating expenses (incl. taxes)	830.5	788.3	668.6
Operating revenues	1,028.3	982.3	840.2
Operating ratio	.81	.80	.80
Retained earnings	426.1	400.9	321.7
Earnings per share of common	$2.47	$2.57	$2.42

	1969		1968	
Capitalization at 12/31	Amount	% of Total	Amount	% of Total
[illegible]-term debt	$1,981.6	51.[illegible]%	$1,901.6	51.9%
[illegible]red stock	626.6	16.4	627.0	17.1
[illegible] stock	1,210.2	31.7	1,139.0	31.0
	$3,818.4	100.0%	$3,667.6	100.0%

Moody's Bond Ratings:	
First Mortgage Bonds	A
Dun and Bradstreet Credit Rating	AaA1

EXHIBIT K

November 16, 1970

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY OF NEW YORK, INCORPORATED

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

BUCHANAN, NEW YORK

DOCKET NO. 50-247

TABLE OF CONTENTS

1.0 INTRODUCTION

The Consolidated Edison Company of New York, Inc., (applicant) filed with the Atomic Energy Commission (AEC or Commission) an application dated October 15, 1968, for an operating license for its Indian Point Nuclear Generating Unit No. 2. Indian Point Unit 2 has been under construction since issuance of a provisional construction permit on October 14, 1966.

Indian Point Unit 2 is located on a 227-acre site on the east bank of the Hudson River at Indian Point, Village of Buchanan, in upper Westchester County, New York.

Indian Point Unit 2 is the first of the four-loop, current generation Westinghouse pressurized water reactor designs. It will be owned and operated by the Consolidated Edison Company of New York, Inc. The Westinghouse Electric Company (Westinghouse) is the principal contractor and has turnkey responsibility for the design, construction, testing, and initial startup of the facility. Westinghouse contracted with United Engineers and Constructors as architect engineer. Construction of the plant was performed by United Engineers until December 1969 when this function was assumed by WEDCO, a wholly-owned subsidiary of Westinghouse.

The operating license application is for a power level of 2758 megawatts thermal (MWt), the same as was requested in the construction permit application. Our evaluation of the engineered safety features (with the exception of the emergency core cooling system) and our accident analyses, have been performed for a maximum power of 3216 MWt.

Our evaluation of the thermal, hydraulic, and nuclear characteristics of the reactor core and the performance of the emergency core cooling system was for a power rating of 2758 MWt. Before operation at any power level above 2758 MWt is authorized, the regulatory staff will perform a safety evaluation to assure that the core can be operated safely at the higher power level.

Our technical safety review of the design of this plant has been based on Amendment No. 9 to the application, the Final Facility Description and Safety Analysis Report (FFDSAR), and Amendments Nos. 10-25, inclusive. All of these documents are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, Washington, D.C. The technical evaluation of the design of this plant was accomplished by the Division of Reactor Licensing with assistance from the Division of Reactor Standards and various consultants to the AEC.

In the course of our review of the application, many meetings were held with representatives of the applicant to discuss the plant design and proposed operation. As a consequence of our review, additional information was requested, which the applicant provided by amendments to the application. A chronology of the principal actions relating to the processing of the application is attached as Appendix A to this safety evaluation. In addition to our review the Advisory Committee on Reactor Safeguards (ACRS) independently reviewed the application and met with both the AEC staff and the applicant on several occassions to discuss the plant. The ACRS report on Indian Point Unit 2, dated September 23, 1970, is attached to this Safety Evaluation as Appendix B. Appendices C through G include reports from our consultants on meteorology, hydrology, seismic and structural design, and radiological monitoring. Appendix H contains the staffs evaluation of the applicant's financial qualifications.

Based upon our evaluation of the plant as summarized in subsequent sections of this report, we have concluded that Indian Point Unit 2 can be operated at thermal power levels of up to 2758 MWt without endangering the health and safety of the public. Subsequent to the issuance of an operating license the unit will be required to operate in accordance with the terms of the operating license and the Commission's regulations under the surveillance of the Commission's regulatory staff.

2.0 FACILITY DESCRIPTION

Indian Point Unit 2 is one of three reactors currently planned for the Indian Point site. Indian Point Unit 2 is adjacent to Indian Point Unit 1, a 615 MWt pressurized water reactor plant that has been in operation since August 1962. Indian Point Unit 3, a plant similar to Indian Point Unit 2, received a provisional construction permit in August 1969, and is presently under construction at the Indian Point site. Each unit has its own auxiliary systems and safety features. The three units, however, will share a common inlet water canal and a common discharge canal. In addition, the controls for Indian Point Unit 2 and Indian Point Unit 1 are located in separate portions of a common control room.

The Indian Point Unit 2 pressurized water reactor is fueled with slightly enriched uranium dioxide in the form of ceramic pellets contained in zircalloy fuel tubes. Water serves as both the moderator and the coolant. Heat is removed from the reactor core by four separate coolant loops, each provided with a separate pump and steam generator. The heated water flows through the steam generators where heat is transferred to the secondary (steam) system. The water then flows back to the pumps to repeat the cycle. The system pressure is controlled by the use of a pressurizer in which steam and water are maintained in thermal equilibrium.

The secondary steam produced in the steam generators is used to drive the turbine generator. The heat of condensing steam is rejected to the circulating water system and discharged to the Hudson River. The condensate is then recharged to the steam generators to repeat the secondary cycle.

The primary coolant system includes the reactor, steam generators, primary coolant pumps, primary coolant piping, and the pressurizer. This system is housed inside the containment building which is a steel-lined, leak-tight reinforced concrete structure. The containment provides a barrier to the release to the environment of radioactive fission products that might be released inside the containment in the event of an accident. Auxiliary systems, including the chemical and volume control systems, the waste handling system, and additional auxiliary cooling systems, are housed separately, principally in the adjacent primary auxiliary building. The primary auxiliary building also houses components of the engineered safety features. A separate fuel handling building is provided for storage of spent fuel. A separate turbine building houses the turbine generator.

Control of the reactor is achieved by reactivity control using top entry control elements that are moved vertically within the core by individual control drives. Boric acid dissolved in the coolant is used as a neutron absorber to provide long-term reactivity control.

To assure reactor operation within established limits, a reactor protection system is provided that automatically initiates appropriate actions whenever plant conditions monitored by the system approach preestablished limits. The reactor protection system acts to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

The engineered safety features include an emergency core cooling system that will cool the reactor core in the event of an accident that results in loss of the normal coolant, containment cooling and iodine removal systems that provide for removal of heat and radioactive iodine from the containment atmosphere should such action be required, and a hydrogen control system that will limit the accumulation of hydrogen within the containment in the event of a loss-of-coolant accident. A containment penetration pressurization system and seal water injection system are provided to assist in isolating the containment in the event of an accident and prevent the escape of fission products to the environment outside the plant.

3.0 SITE AND ENVIRONMENT

3.1 Population and Land Use

The Indian Point site consists of 227 acres in the town of Buchanan in upper Westchester County, New York, approximately 24 miles north of the New York City boundary line. The estimated population distribution in the vicinity of the site is presented in table 2.1.

TABLE 2.1

CUMULATIVE POPULATION

Distance (miles)	1960 (U.S. Census)	1980 (Projected)
0-1	1,080	2,100
0-2	10,810	20,900
0-3	29,630	59,520
0-4	38,730	78,800
0-5	53,040	108,060
0-10	155,510	312,640

The minimum radius of the exclusion area* for Indian Point Unit 2 is 520 meters. The applicant has chosen 1100 meters as the outer

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*Exclusion area is defined in the Commission's Site Criteria, 10 CFR Part 100, as that area surrounding the reactor in which the reactor licensee has the authority to determine all activities including removal of personnel and property from the area.

boundary of the low population zone** because of the limited population within this distance from the plant.

The Commission's site criteria guidelines state that the population center distance*** should be at least 1-1/3 times the distance from the reactor to the outer boundary of the low population zone (LPZ), but also state that in applying this guide due consideration should be given to the population distribution within the population center. The nearest corporate boundary of Peekskill (population 19,000) is approximately 800 meters (0.5 miles) from Indian Point Unit 2. Because of the limited population within the low population zone (66) including that portion of Peekskill within the zone, and because Peekskill is of a generally industrial nature in the vicinity of the site and the resident population within and out to 1-1/3 times the low population zone distance is low, we concluded at the time of our construction permit review that the distance selected by the applicant for the exclusion area radius, the LPZ outer boundary, and the population center distance meet the intent of the 10 CFR Part 100 guidelines and are acceptable. On the basis of our evaluation of the potential radiological consequences of postulated design basis accidents,

**Low population zone is defined in the Commission's Site Criteria, 10 CFR Part 100, as the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident.

***Population center distance is defined in the Commissions Site Criteria, 10 CFR Part 100, as the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

we conclude that the calculated doses presented in Section 11.0 of this evaluation are well within the guidelines of 10 CFR Part 100 for these distances.

3.2 Meteorology

The meteorology of the Indian Point site is affected by its position in a deep river valley. Consequently, the wind direction generally follows a pronounced diurnal cycle with unstable flow in the up-river direction during the daytime and stable flow in the down-river direction at night.

The applicant has presented the results of meteorological measurements taken at the site over a period of two years including windspeed, wind direction, and temperature lapse rate data for various heights. We have reviewed the data presented and conclude that they provide an adequate basis for selecting the meteorological parameters used in determining the routine effluent release limits and in evaluating the consequences of postulated accidents. The comments of our meteorological consultants, the Environmental Science Service Administration (ESSA) support this conclusion and are attached as Appendix C.

3.3 Geology and Seismology

During our review of this site prior to issuance of the construction permit for Indian Point Unit 2, we and our consultant, the U. S. Geological Survey, concluded that the geology of the site provides an adequate founding medium for the plant buildings and structures. No new developments have occurred during the construction permit review of Indian Point Unit 3 or otherwise since our construction permit review for Indian Point Unit 2 to change our previous conclusion on the acceptability of the geological and seismological features of the Indian Point site.

Maximum ground accelerations of 0.10g and 0.15g were used for the Operating Basis Earthquake* and the Design Basis Earthquake**, respectively. These values were selected at the time of the construction permit review. At that time we and our consultant, the U. S. Coast and Geodetic Survey, concluded that they were acceptable for the site.

A strong motion seismograph has been installed on a concrete slab directly on bedrock in the yard area of the plant to record data related to ground motion in the event of a seismic disturbance at or near the site. These data would be employed in an evaluation of the effects of the seismic disturbance to assure the capability for continued safe operation of the plant.

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*"Operating Basis Earthquake" for a reactor site is that earthquake which produces vibratory ground motion for which those structures, systems and components, necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

**"Design Basis Earthquake" for a reactor site is that earthquake which produces vibratory ground motion for which those structures, systems, and components, necessary to shut down the reactor and maintain the unit in a safe shutdown condition without undue risk to the health and safety of the public are designed to remain functional.

3.4 Hydrology

The applicant has reevaluated the potential flooding that could occur at the site. The following hypothetical flood conditions were analyzed: (1) the probable maximum flood peak discharge of 1,100,000 cubic feet per second resulting from the probable maximum precipitation occurring over the total basin, a 12,650 square mile area above the plant site; (2) the flooding caused by failure of the Ashoken Dam concurrent with a major river basin flood (standard project flood) with a peak discharge of 705,000 cubic feet per second and a hurricane storm surge (standard project hurricane), and (3) the probable maximum hurricane concurrent with the high spring tide in the Hudson River. These three hypothetical floods are the most severe of several investigated, and each of the three results in a maximum water surface elevation of about 15 feet above mean sea level. We have reviewed the method of calculation and conditions assumed and find that they are conservative and acceptable. Both the U. S. Geological Survey and the Coastal Engineering Research Center provided consulting services with respect to our flooding evaluation. Their reports are attached as Appendix D and Appendix E, respectively.

3.5 Environmental Monitoring

The radioactivity levels in the vicinity of the Indian Point site have been measured by the applicant since 1958 to ascertain the impact of operation of Indian Point Unit 1 on the background levels of radioactivity. The environs of the Indian Point site have been studied intensively for many years by the Institute of Environmental Medicine at New York University Medical Center. These studies concerned both the exposure to man and to the flora and fauna indigenous to the Hudson River. All the results compiled to date indicate that radioactive effluents from Indian Point Unit 1 operation have produced barely quantifiable radiation exposure to the public and have had no detectable effect on the ecology of the area.

The operational environmental radiation monitoring program for Indian Point Unit 2 will be a continuation of the present program. The program includes direct measurements of gamma radiation and analyses to monitor fallout, air particulates, airborne iodines, water from various surface drinking water supplies, Hudson River water, water from lakes near the site, well water, lake aquatic vegetation, Hudson River vegetation, river bottom sediments, river aquatic biota, terrestrial vegetation, and soil. The report of the U. S. Department of the Interior is attached as Appendix G. This report incorporates the comments of the Federal Water Quality Administration, the Fish and Wildlife Service, and the Bureau of Outdoor Recreation. The report comments favorably on current activities being performed by or for the applicant in connection with determining the effects of both radiological and thermal discharges at the plant site. Recommendations for continued effort in the area of environmental monitoring and ecological studies are included in the report. This report has been forwarded to the applicant.

We conclude that the applicant's program will be adequate for monitoring the radiological effects of Indian Point Unit 2 operations on the environment and for assessing the effects of releases of radioactivity to the environment from operation of the plant on the health and safety of the public.

4.0 REACTOR DESIGN

4.1 General

The nuclear reactor for Indian Point Unit 2 was designed and manufactured by Westinghouse. The principal design features, materials of construction, and arrangement of various components of the Indian Point Unit 2 core are the same as those for the Rochester Gas and Electric Company's R. E. Ginna facility (Docket No. 50-244), which has been licensed for operation by the Commission and which has completed almost a full year of power operation. Further, the zircalloy clad fuel, burnable poison in the initial core loading, a chemical neutron absorber, and part-length control rods to shape axial power distribution are used in substantially the same manner in both the Ginna and the Indian Point Unit 2 reactors. On the basis of our previous review of all of these features for the Ginna reactor, we conclude that these same features are acceptable for Indian Point Unit 2.

4.2 Nuclear Design

The Indian Point Unit 2 reactor core differs principally from the Ginna and Connecticut Yankee (Docket No. 50-213) reactor cores in that the Indian Point Unit 2 reactor core is somewhat larger. The Indian Point Unit 2 core is about 23% greater in cross sectional area and 20% longer than the Connecticut Yankee core and about 89% greater in cross sectional area and the same length as the Ginna core. Because this larger core could be subject to power oscillations or power tilts, we reviewed the nuclear design and power distribution detection and control systems for the Indian Point Unit 2 reactor core in detail.

During plant operation, changes in the core power level or the control rod configuration can cause time-dependent variations in the local power distribution as a result of variations in the concentration of fission products and their radioactive decay products. The most significant fission product-decay product chain with regard to core behavior is the decay of iodine-135 to xenon-135 since the latter is a strong absorber of thermal neutrons. The local oscillations in the neutron flux and in the power level can occur even though the average power level of the core is maintained constant, and the magnitude of the oscillations may decrease, remain constant, or increase with time.

The spatial stability of the xenon distribution and resultant core power peaking abnormalities for the Indian Point Unit 2 core have been investigated by Westinghouse with the conclusion that the core is stable against various types of xenon induced spatial oscillations in the X,Y horizontal plane. This conclusion is supported by analysis and by experiments performed in the Connecticut Yankee reactor. An initial test program for Indian Point Unit 2 will be performed to verify this stability. If this initial test program does not demonstrate stability, the applicant has agreed to operate with partially inserted control rods, or to add fixed or burnable poison shims sufficient to assure stability lead to planes of weakness and cause cracking under conditions of over-loading. The pressure tests, however, will reveal any such cracking.

Approximately one in 200 splices was removed for test purposes. This is generally adequate.

Instrumentation and Controls

At the time of the May 1969 visit it was ascertained that the applicant considers the control room as a Class I structure and intends that the housing of it will also be subject to Class I requirements. However, the instrumentation for the control room as well as other instrumentation critical to containment and safe shutdown, has been purchased from the vendors according to applicant's specifications. The answer to Question 1.9 describes the vibration tests employed for selected items of essential equipment; the purpose of these tests is to help demonstrate that little or no difficulty will be expected in the operating characteristics thereof under seismic conditions. Although not absolute proof of acceptability, satisfactory test results certainly help to confirm the adequacy of such instrumentation and control items. Further information on the design and procurement approach for protection system equipment is given in the answer to Question 7.27 (Suppl. 13), and lends confirmation to the approach adopted.

Tornado Loadings

The information contained in Section 3.4 of the containment design report, and the answer to Question 5.7 of the FSAR indicates that the structure is designed for the usual wind loadings. The analyses described in Appendix B of Supplement 6, indicate that the containment building can resist the design tornado. What effect if any that a tornado could have on the control room or other critical facilities is not stated. However, the applicant states that through reduction of the moderator temperature coefficient, or to operate at reduced power levels. Because of the test program that will be performed and the operating limitations that will be imposed if required, we conclude that the reactor will be stable with respect to potential power oscillations in the X,Y horizontal plane.

The analysis made by Westinghouse indicates that the reactor may be subject to divergent xenon oscillations in the axial direction, resulting in an axial power distribution imbalance or tilt. In view of this, it is assumed that the axial power tilts can occur, and provision is made to detect and control differences in the fraction of the total power generated in the upper and lower halves of the core. Data correlations have been made at the Connecticut Yankee reactor and at the Ginna reactor to relate the readings obtained from the split out-of-core detectors to axial power tilts. Additional correlations will be established during the Indian Point Unit 2 startup tests. Part-length control rods are provided to prevent unacceptable axial power tilts and to control potentially divergent axial xenon spatial oscillations. Analytical studies and experience with the Ginna reactor, provide assurance that any axial oscillations can be controlled such that the power distribution will be maintained within design limits. In addition, automatic protective action is provided to avoid exceeding design power peaking factors at full power in the event of control system malfunctions. To accomplish this, the overtemperature ΔT and overpower ΔT trip set points are automatically reduced in proportion to the axial power tilt as measured by the split out-of-core neutron detectors. We conclude that the system of detection instrumentation, control with part length rods, and automatic protection for potential axial power tilts is acceptable.

Even in the absence of xenon induced instability, power tilts or imbalances can occur in the horizontal or axial planes as a result of control rod misalignment. Analyses for Indian Point Unit 2 and experiments in the Connecticut Yankee reactor have shown that these power tilts can be detected by (1) the split out-of-core neutron detectors, (2) the core exit thermocouples, or (3) the movable in-core neutron detectors. All of these detectors are required to be operable by the Technical Specifications. In addition detection will ordinarily be readily accomplished by the fixed in-core neutron instrumentation.

The power distribution in the Indian Point Unit 2 core is expected to be stable or only slowly varying within known limits and adequate core instrumentation will always be available to detect, monitor, and diagnose any significant power maldistributions.

We conclude that the Indian Point Unit 2 reactor core nuclear design and instrumentation is acceptable.

4.2 Thermal-Hydraulic Design

We have evaluated the adequacy of the core thermal and hydraulic design, both for steady-state plant operation and for anticipated plant transients. The design criteria selected by the applicant to prevent fuel damage are: (1) the departure from nucleate boiling (DNB) ratio (determined using the Westinghouse W-3 correlation) shall not be less than 1.3 during normal plant operation or as a result of anticipated transients; and (2) no fuel melting shall occur during either normal operation or anticipated transient conditions. The anticipated plant transients that result in the most severe core thermal transients are loss of coolant flow, excessive load increase, and a loss of external electrical load. The applicant's analyses show that the DNB ratio will be greater than 1.3 for each of these plant transients when operating at the license power level of 2758 MWt. The lowest DNB ratio calculated as a result of any of the plant transients, was for the case of simultaneous loss of electrical power to the four reactor coolant pumps. This transient results in a DNB ratio of 1.42. In addition, no fuel melting is predicted to occur for steady-state operation or as a result of anticipated transients.

As stated above the Indian Point Unit 2 reactor core is designed to undergo anticipated plant transients with a minimum DNB ratio greater than 1.3. On this basis, clad temperature should not be significantly affected by a transient and no fuel failure should occur for the range of fuel element burnup planned for the Indian Point Unit 2 core. As part of a continuing experimental effort to demonstrate satisfactory performance of fuel at high burnup and high power density, Westinghouse is continuing a fuel irradiation program at conditions significantly in excess of current PWR design limits, and will establish power burnup limits for the fuel. These irradiation programs are being conducted at both the Saxton and Zorita reactors. Sustained operation of selected fuel rods at peak design power levels in the Zorita reactor will increase assurance that the fuel has adequate margins to accommodate transient overpower operation.

Based on our evaluation of the results of these analyses, and on our review of the design limits and the operating experience of similar reactors, we conclude that the reactor core thermal and hydraulic design is acceptable for operation at the rated power of 2758 MWt.

5.0 REACTOR COOLANT SYSTEM

5.1 General

The reactor primary coolant system, including all vessels, pumps, and piping is designed for a pressure of 2485 psig and a temperature of 650°F. The system has been designed to withstand, within the stress limits of the codes used in the design, the normal loads of mechanical, hydraulic, and thermal origin, plus those due to anticipated transients and the operating basis earthquake.

5.2 Primary System Components

The reactor internals are designed to withstand the normal design loads of mechanical, hydraulic, and thermal origin, including those resulting from anticipated plant transients and the operating basis earthquake, within the stress limit criteria of Article 4 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III. Although the Indian Point Unit 2 reactor internals are not designed to withstand simultaneously the loads resulting from loss-of-coolant accident blowdown and seismic events, the applicant has submitted a summary of an analytical study of the behavior of the reactor internals under simultaneous blowdown and seismic loadings (WCAP-7332-L). The results of this study indicate that for the combined blowdown and design basis earthquake loadings the resulting deflections are within the loss-of-function limits except for the control rod immediately adjacent to the coolant line that was assumed to fail. On the basis that the core reactor internals remain functional and that adequate shut down margin can be achieved by control rod insertion, we conclude that the stress and deflection limits for the combined blowdown and design basis earthquake loadings provide an adequate margin of safety.

The primary system side of the steam generators, the pressurizer, and the main coolant pump casings, have been designed to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition - Summer 1969 Addenda, as Class A vessels. For other Class I pumps, valves, and heat exchangers the inspection program required independent review of (1) the physical and chemical test data for pressure boundary materials, (2) radiographs of valve bodies, valve bonnets and pump casings, and (3) dye-penetrant examinations of heat exchanger tubes and welds. These requirements resulted in fabrication and inspection programs that contain the essential elements of the recently proposed ASME Codes for Nuclear Pumps and Valves. We find the design codes and inspection requirements acceptable.

We have reviewed the information submitted by the applicant with respect to operating limitations on heatup and cooldown of the primary system imposed by the fracture toughness properties of the materials of the Indian Point Unit 2 reactor vessel. Our evaluation was based on a proposed redraft of section NB-2300 Special Materials Testing (Section III ASME Boiler and Pressure

Vessel Code) dated July 28, 1970, which reflects the material testing requirements in a form consistent with the AEC Fracture Toughness Criteria. As a consequence of our evaluation the applicant has agreed to the heatup and cooldown limitation as presented in Section 3.1-B of the Technical Specifications which represents a modification of his initial submittal. On the basis that these limits reflect a very conservative method of defining pressure vessel fracture toughness, we conclude that they are acceptable.

5.3 Coolant Piping

The reactor coolant piping has been designed in accordance with the requirements of the American National Standards Institute (ANSI) B31.1 Code for Power Piping, 1955 Edition, including the requirements of Nuclear Code Cases N-7 and N-10. All welding procedures and operators were qualified to the requirements of Section IX of the ASME Boiler and Pressure Vessel Code. Additional inspection requirements for the reactor coolant piping during fabrication included ultrasonic and dye-penetrant inspection of all pipe welds. Non-destructive examination of valves included radiographic examination of the valve castings and ultrasonic inspection of all forged components. Dye-penetrant surface examination was also performed. With this program, the inspection of the Indian Point Unit 2 reactor coolant piping substantially meets the requirements of Class 1 systems under the ANSI B31.7 Code for Nuclear Power Piping adopted in 1969. On this basis we have concluded that the design and inspection program for this system is acceptable.

The original seismic design analysis for the Indian Point Unit 2 reactor coolant system utilized only static methods of analysis. Recently, at our request, the applicant completed a rigorous dynamic analysis of this system utilizing both modal-response spectra and model time-history methods of analyses. As with the reactor internals, the combined loading of a concurrent loss-of-coolant accident blowdown and design basis earthquake was not considered in the design of the Indian Point Unit 2 reactor coolant system. However, the applicant recently completed an analysis of the response of the reactor coolant system to be installed in Indian Point Unit 3 for these combined loads. Since the Indian Point Unit 3 and the Indian Point Unit 2 reactor coolant systems are identical, the applicant has used the results of the analysis for Indian Point Unit 3 in conjunction with the material properties for the Indian Point Unit 2 piping, as determined from tests, to determine that the combined seismic and accident loads can be tolerated by the Indian Point Unit 2 reactor coolant system within acceptable stress limits.

Based on our review of the design limits and analytical procedures employed, we find that the design of the Indian Point Unit 2 reactor coolant system is acceptable.

5.4 Other Class I* (Seismic) Piping

At our request the applicant performed additional seismic analysis on other Class I piping. The adequacy of the seismic design of the feedwater lines, pressurizer surge line, and a typical steam line has been confirmed by a dynamic analysis utilizing the modal-response-spectra method. The adequacy of the seismic design of other Class I (Seismic) piping in the plant was determined by performing a dynamic analysis on selected "worst case" systems. Several systems that are the most vulnerable to dynamic excitation because of system flexibility or location in the supporting structure were analyzed and the resulting stresses compared with the stresses determined by the original static analyses. The applicant has concluded that the conservatism of the original static analysis provided adequate margins to accommodate the previously undetermined dynamic effects.

Based on our review of the original static methods employed and the confirmatory evidence obtained from the recent dynamic analyses of the most vulnerable systems, we have concluded that the design of the Class I (Seismic) piping systems in Indian Point Unit 2 is acceptable.

*See Section 6.1 for definition of Class I structures, systems, and components.

5.5 Inservice Inspection

An inservice inspection program for the reactor coolant system is included in the Technical Specifications. This program follows Section XI of the ASME Code, Rules for Inservice Inspection of the Reactor Coolant System, as closely as practical. The design of the primary system including the capability to remove insulation at selected areas provides an acceptable degree of access for inspection purposes. The applicant also intends to conduct periodic inservice inspections of the primary pump motor flywheels.

The applicant will review the inservice inspection program with us after five years of reactor operation. It may then be modified based on experience gained during these five years. At that time, we will also require the applicant to perform such inspections of components outside the reactor coolant pressure boundary as deemed necessary to provide continuing assurance of structural integrity.

5.6 Missile Protection

We have reviewed the applicant's primary system layout within the containment in terms of the protection afforded the containment liner and Class I (seismic) systems inside the containment from missiles that might be generated as a result of a primary system failure. We have concluded that adequate protection from potential missiles is provided by the system arrangement and surrounding thick circumferential concrete walls and the concrete floors.

The primary pump motor flywheels installed in Indian Point Unit 2 are the same as those in use in other plants. The flywheels are the standard Westinghouse design, fabricated of A 533B steel. On the basis of the use of high grade material, extensive quality control measures, special manufacturing procedures and preservice and inservice surveillance requirements, we have concluded that assurance has been provided that the integrity of the flywheels will be maintained.

5.7 Leak Detection

The reactor coolant pressure boundary leak detection systems for this plant are similar to those we have reviewed and found acceptable for other plants using a Westinghouse nuclear steam supply system. The systems are based upon air particulate monitoring, radiogas monitoring, humidity detection, and containment sump level monitoring. These systems provide an array of instrumentation that is sensitive, redundant, and diverse and that has adequate alarm features. The sensitivity of these systems is consistent with their primary purpose of detecting any leak in the primary system pressure boundary which could be indicative of incipient failure. The Technical Specifications require that two reactor coolant leak detection systems of different principles shall be in operation when the reactor is operated at power. We conclude that the leak detection systems for Indian Point Unit 2 are acceptable.

5.8 Fuel Failure Detection

The fuel element failure detection system will measure delayed neutron activity in one hot leg of the reactor coolant system. The monitor is connected in series with a delay coil to allow a decay time for N^{16} gamma activity (half life of 7.1 seconds) of about 60 seconds before the coolant reaches the detector. This delay reduces gamma ray background and facilitates detector sensitivity. An alarm signal is provided for the channel. We conclude that this system which is inherently faster in response than previous systems reviewed for other reactors is acceptable.

5.9 Vibration Monitoring and Loose Parts Detection

The major core and core support components have been analyzed to provide assurance that they are not vulnerable to vibratory excitation. Vibration analyses for the core support barrel considered inlet flow impingement and turbulent flow. Natural frequency calculations were made to assure that there would be no deleterious response to known excitations such as pump blade passing and driven frequencies. Fuel bundle response to anticipated driving forces has been calculated and determined by tests in the Westinghouse Reactor Evaluation Center.

The vibration monitoring system to be used for the preoperational test program on Indian Point Unit 2 will consist of mechanical gauges to measure gross relative motion between the thermal shield and core barrel, strain gauges on selected guide tubes, and accelerometers on the upper core plate. We have concluded that the vibration design analyses and the preoperational test program are acceptable.

In the course of our review of the Indian Point Unit 2 application, it has been noted that techniques for the analysis of neutron noise spectra and accelerometer measurements on the lower heads of primary system vessels might be developed to provide a useful method for inservice monitoring of reactor coolant systems to detect changes in the vibration of reactor components or the presence of loose parts. The applicant has stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefullness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

5.10 Conclusion

Based on our review of (1) the codes and standards used for design, (2) the fabrication and inspection procedures, (3) the inservice inspection program, (4) the provisions for missile protection and leak detection, (5) the provision for fuel failure detection, and (6) the provisions for preoperational vibration testing and the developmental effort for inservice monitoring to detect vibrations and loose parts, we have concluded that the design and inspection procedures for the reactor coolant system for the Indian Point Unit 2 are acceptable.

6.0 CONTAINMENT AND CLASS I (SEISMIC) STRUCTURES

6.1 General Structural Design

The applicant has categorized as Class I (seismic) those structures (e.g., containment structure and primary auxiliary building), and those systems and components (e.g., reactor vessel and internals, emergency core cooling system), whose failure could cause a significant release of radioactivity or that are vital to the safe shutdown of the facility and the removal of decay heat. We have reviewed the applicant's classification of structures, systems, and components and conclude that they have been classified appropriately.

The Class I (seismic) structures at Indian Point Unit 2 are the containment structure, the primary auxiliary building, the control room building, the fuel storage pool, the diesel generator building, and the intake structure and service water screenwell. The major portion of the primary auxiliary building, the fuel storage pool, and the intake structure are of reinforced concrete construction. The control room building, the diesel generator building, the fuel storage building and the non-Class I portions of the primary auxiliary building are constructed of steel framing with composite metal panel siding.

The environmental conditions that were considered in the structural design include the operating basis earthquake (OBE), the design basis earthquake (DBE), the flooding and wind due to the probable maximum hurricane, and the flooding due to the probable maximum flood. We have concluded that these conditions were used for the design in an acceptable manner.

6.2 Structural Design and Analysis

The Indian Point Unit 2 primary containment has a free volume of 2.6 x 10^6 cubic feet and a design pressure of 47 psig. The containment structure is a right cylinder (thickness 4.5 ft) with hemispherical dome (thickness 3.5 ft) mounted on a flat (thickness 9 ft) base mat. The reinforced concrete is lined with 1/4 inch minimum thickness welded ASTM A442 grade 60 firebox quality carbon steel plate. The reinforcing bars conform to ASTM A432 specifications. The reinforcing in the cylinder wall is placed in horizontal and vertical directions with added diagonal tangential reinforcing for earthquake resistance. The reinforcing bars conform to ASTM A432 specifications. Cadweld splices are used in 14S and 18S bars.

We have evaluated the pressure transients that might occur in the containment in the event of a loss-of-coolant accident assuming various sizes of primary coolant system breaks. For the range of postulated break sizes up to and including the double-ended severance of the largest reactor coolant pipe, the largest calculated peak containment pressure is 40 psig. The design pressure of the containment exceeds the calculated peak pressure by more than 10% and is acceptable.

The containment is designed to remain within the elastic range for the 0.10g OBE concurrent with the accident and other applicable loads. It is also designed to withstand the 0.15g DBE concurrent with the accident without loss of function.

We and our seismic design consultant, Nathan M. Newmark, are in agreement with the loading combinations and allowable stresses used by the applicant. Stress and strain limits conform to the requirements of ACI 318-63, Part IV-B. The ACI load factors have been replaced by factors suitable for concrete containment structures.

Based on our review of the design of the containment structure and its capability to withstand the predicted pressures from potential accidents, we conclude that the structural design aspects of the containment are acceptable.

In evaluating the capability of the Class I (seismic) structures, systems, and components, to withstand the dynamic loads due to seismic events, our seismic design consultant, Nathan M. Newmark Consultant Engineering Services, considered the geology and nature of the bedrock, design loads and load combinations, the seismic design parameters, and methods of analysis. On the basis of our review and that of our seismic design consultant, we conclude that the Class I (seismic) structures, systems, and components of Indian Point Unit 2 are designed to accommodate all applicable loads and are acceptable. The report of our seismic design consultant is attached as Appendix G.

During our review we noted a limited number of cases where failure of non-Class I (seismic) structures could potentially endanger Class I (seismic) structures and equipment. These included the Indian Point Unit 1 superheater stack and superheater building, the turbine building, and the fuel storage building. In response to our concern, the applicant performed analyses of these structures using a multi-degree of freedom modal dynamic analysis method, to determine the modifications needed to assure that gross structural collapse of these structures would not occur in the event of a DBE. As a result of these analyses, additional seismic reinforcement is being provided for both the superheater building and the turbine building and the Indian Point Unit 1 superheater stack is to be reduced in height by 80 feet. The truncation of the stack is to be accomplished at a convenient time in the next three years and prior to operation of Indian Point Unit 3. We and our seismic design consultant have reviewed the material submitted by the applicant and conclude that the dynamic analyses performed, and the design modifications proposed, are acceptable.

We have reviewed the as-built wind resistance of Class I structures at the Indian Point Unit 2 facility. Analysis indicates that both the containment and reinforced concrete portions of the primary auxiliary building and intake structure can sustain winds in the range of 300 miles per hour. The control building and diesel generator building which are constructed of structural steel with composite metal panel siding, are estimated by the applicant to be capable of sustaining wind loads of up to 160 miles per hour.

is being installed on the containment for strength testing, although examinations will be made for cracking and distortion during the pressure test. Periodic leakage rate tests will be performed on the containment and its penetrations.

We have concluded that the provisions for testing and surveillance of the containment are acceptable. Test and surveillance requirements are included in the Technical Specifications.

6.4 Missile Protection

The possibility exists that missiles might be generated in the unlikely event of a failure of the turbine generator. Although the design criteria for Indian Point Unit 2 did not include consideration of protection against missiles resulting from turbine failures, at our request the applicant has assessed the protection available against missiles that might result from a turbine failure at the maximum overspeed condition (133% of rated normal speed). Specific provisions have been added to limit the off-site consequences that could result from a missile failure, and to provide for safe shut down of the unit. These include an alternative cooling water supply for the charging pumps and added missile protection for a potentially vulnerable portion of the auxiliary steam generator feedwater lines. In addition, a second completely independent turbine speed control system has been provided to reduce the probability of a runaway speed condition that might result in a turbine failure. This system is designed to the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Criteria No. 279 for protection systems. The Technical Specifications require periodic testing of the overspeed devices to assure operability. We conclude that the applicant has made appropriate provisions to reduce the probability of a destructive turbine missile from being generated and affecting Class I (seismic) items.

The Indian Point Unit 2 reactor vessel cavity is designed to protect the containment against missiles that might be produced by postulated failure of the reactor vessel. Failure of the reactor vessel would result in fluid jet-reaction forces in the cavity wall adjacent to the vessel split or crack as well as stress in the cavity wall from a rise in cavity pressure, both of which would result from coolant blowdown. Also reaction forces in the cavity wall and floor might be produced by the impact of missiles generated by pressure vessel failure. By the use of extensive steel reinforcing, the concrete cavity has been designed to resist both fluid jet and missile impact forces that could result from pressure vessel failure by either longitudinal splitting or various modes of circumferential cracking. The cavity is also designed to sustain a fluid pressure rise to 1000 pounds per square inch. We have reviewed the applicant's analysis and conclude that the cavity as designed provides adequate protection for the containment liner against missiles that might result from a postulated pressure vessel failure.

7.0 ENGINEERED SAFETY FEATURES

7.1 Emergency Core Cooling System

The principal equipment of the emergency core cooling system consists of (1) three 50% capacity high pressure safety injection pumps, (2) two 100% capacity residual heat removal pumps for low pressure injection and external recirculation, (3) two 100% capacity recirculation pumps for recirculation internal to the containment, (4) one 100% capacity boron injection tank, and (5) four 33-1/3% capacity accumulators. This system provides redundant capability to inject borated cooling water rapidly into the core in the event of a loss-of-coolant accident and to maintain coolant above the level of the core for an indefinite period following the accident.

The applicant's evaluation of the performance of these systems is based on detailed analyses of (1) the hydraulic behavior of the primary coolant system during and subsequent to a loss-of-coolant accident, and (2) the thermal response of the core during the same period. The analytical methods used to predict the hydraulic behavior of the primary coolant system during a loss-of-coolant accident have been improved significantly during the construction period for Indian Point Unit 2. The original analysis presented in Volume 4 of the FFDSAR was performed with the FLASH-1 hydraulics computer program. This program is limited to a three-node

We conclude that the emergency core cooling system will (1) limit the peak clad temperature to well below the clad melting temperature, (2) limit the fuel clad water reaction to less than 1% of the total clad mass, (3) terminate the clad temperature transient before the geometry necessary for cooling is lost and before the clad is so embrittled as to fail upon quenching and (4) reduce the core temperature and then maintain core and coolant temperature levels in a subcooled condition until accident recovery operations can be accomplished.

In summary, we conclude that the emergency core cooling system is acceptable and will provide adequate protection for any loss-of-coolant accident.

The emergency core cooling system design as presently installed at Indian Point Unit 2 was reviewed by the Division of Reactor Licensing during 1967, subsequent to the issuance of the construction permit on October 14, 1966. This system represented a complete redesign, a considerable increase in flow capability, and enhanced performance when compared to the system reviewed for the construction permit. On the basis that the very significantly improved performance of the redesigned emergency core cooling system provides additional assurance for limiting clad temperatures and maintaining a coolable core we concurred with the applicant's decision to remove the reactor pit crucible from the facility design.

7.2 Containment Spray and Cooling Systems

Two independent heat removal systems are provided to control the containment pressure and temperature following a loss-of-coolant accident. Each system, acting alone at its rated capacity, will prevent over-pressurization of the containment structure. The two systems are the containment spray system and the fan cooling system. The design of each is substantially the same as the design of systems provided at the Ginna plant and other licensed plants.

The containment spray system consists of two 50% capacity spray pumps and is sized to limit the containment post-accident pressure to below design pressure. Sodium hydroxide and boric acid are used as additives to the spray solution to remove radioactive iodine which might be present in the containment after an accident. We have reviewed the use of these chemical spray additives in terms of their iodine removal capabilities, and in addition have evaluated the chemical compatibility of the spray solution with other reactor components. As a result of our review, we conclude that the spray system is adequately sized to cool the containment, that the alkaline spray solution will reduce the iodine concentration in the containment atmosphere, and that corrosion of other materials used in the containment does not introduce a safety problem.

The containment fan cooling system provides complete redundancy to the containment spray system for heat removal from the containment atmosphere during post-accident conditions. Five 20% capacity fan coolers are provided. Since the fan coolers are located within containment, they must be capable of operating in the post-accident environment. Westinghouse has conducted an environmental test program to demonstrate this capability. Our evaluation of these tests, including the heat removal capability of the heat exchangers, and environmental and radiation testing of the fan cooler motors, valve motor operators and electric cabling indicates that these components will function satisfactorily in the accident environment. An iodine-impregnated charcoal filter system has been included with the fan cooler system to remove organic iodine from the post loss-of-coolant containment atmosphere. The charcoal beds are preceded by demisters and high efficiency particulate air (HEPA) filters.

We have evaluated the inorganic and organic iodine removal capability of the charcoal beds on the basis of tests with steam - air mixtures at 100% relative humidity following prolonged flooding of the bed. We conclude that inorganic and organic iodine removal efficiencies of 90% and 10% per pass, respectively, are conservative values that are justified by the available information.

In summary, we have reviewed the containment spray and fan cooling systems in terms of (1) capability to control the containment temperature, (2) capability to remove inorganic and organic iodine,

(3) system and component redundancy, and (4) capability to function in the post-accident containment environment. We conclude that there is reasonable assurance that these systems will operate as proposed subsequent to a loss-of-coolant accident.

7.3 Containment Isolation Systems

In addition to the usual capability of isolating all lines leading to and from the containment, the Indian Point Unit 2 containment is provided with additional systems to minimize the potential leakage of fission products subsequent to an accident. A containment penetration and weld-channel pressurization system provides for continuous pressurization of zones enclosing containment penetrations and the welds in the containment liner. The system continuously maintains an overpressure of clean, dry air that is in excess of the containment design pressure. Pressurized zones include each piping penetration, each electrical penetration, double gasketed spaces on the personnel and equipment hatches, and the channels over weld seams of the containment liner. The air pressure is maintained by the instrument air compressors with backup from the plant air compressors and from a standby source of nitrogen cylinders. Pressure indication and alarm instrumentation is provided locally and in the control room to assure that loss of pressure will be detected and corrected.

In addition, an isolation seal water system has been provided to assure containment isolation by (1) injecting seal water between the seats and stem packing of the globe and double disc isolation valves used on larger lines, and (2) injecting seal water directly into the line between the closed diaphragm valves used in the smaller lines penetrating containment. Seal water injection is provided for all lines connected to the reactor coolant system and for lines that may be exposed to the containment atmosphere subsequent to an accident. Although the use of the seal water system following a loss-of-coolant accident provides an additional means of reducing leakage, we have not considered the effect of this system in determining the offsite radiological consequences.

We have concluded that the capability provided for isolating the containment is acceptable.

7.4 Post-Accident Hydrogen Control System

In the event of a loss-of-coolant accident, radiation from the core and from escaped fission products will dissociate some of the cooling water into gaseous hydrogen and oxygen. Continued evolution of hydrogen would increase the concentration in the containment to a point where ignition could occur and thus provide an additional energy source.

Redundant flame recombiner units are installed within the Indian Point Unit 2 containment. Each unit has the design capability to prevent the ambient containment hydrogen concentration from exceeding two percent by volume. The units are designed to function, following the loss-of-coolant accident in a containment pressure environment of 1 to 5 psig. Each recombiner system consists of (1) a flame recombiner unit located within containment, (2) a control panel located outside of containment, and (3) a hydrogen gas stand located outside of containment. On the basis of (1) our detailed review of the design of the system and its controls, (2) satisfactory performance testing of the device, and (3) satisfactory environmental testing of those portions of the recombiner system installed within the containment, we conclude that there is reasonable assurance that the recombiner system will perform its intended post-accident function.

In addition, the applicant will provide the capability for purging the containment atmosphere through appropriate filters as an alternate backup means of hydrogen control. The containment penetrations to be used for this system are installed. The design and installation of the equipment required will be performed during the first two years of operation at power.

8.0 INSTRUMENTATION, CONTROL, AND POWER SYSTEMS

8.1 Reactor Protection and Control System

The reactor protection system instrumentation for Indian Point Unit 2 is the same as that installed at the Ginna plant. The adequacy of the protection system instrumentation was evaluated by comparison with the Commission's proposed general design criteria published on July 11, 1967, and the proposed IEEE criteria for nuclear power plant protection system (IEEE-279 Code), dated August 28, 1968. The basic design has been reviewed extensively in the past and we conclude that the design for Indian Point 2 is acceptable.

During our review we considered the adequacy of reactor protection for operation with less than four coolant loops in service. When operating with one of the primary loops out of service the reactor is normally automatically limited to 60% of full power. However by manual adjustment of several protection system set points in a manner consistent with the Technical Specifications adequate reactor protection can be provided for operation up to 75% of full power.

We have reviewed the applicant's analysis of the seismic response of the protection system instrumentation and associated electrical equipment and find that adequate testing has been performed on the nuclear instrumentation, switch gear, and process system instrumentation.

In connection with our review of potential common mode failures we have recently considered the need for means of preventing common failure modes from negating scram action and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has been responsive to our request for information and has provided the results of analyses which indicate that the consequences of such transients are tolerable for the existing Indian Point Unit 2 design at a power level of 2758 MWt. Although additional study is required of this general question, we conclude that it is acceptable for the Indian Point Unit 2 reactor to operate at a power level of 2758 MWt while final resolution of this matter is made on a reasonable time scale.

8.2 Initiation and Control of Engineered Safety Features

The instrumentation for initiation and control of engineered safety features for the Indian Point Unit 2 is the same as that installed at the Ginna plant. This basic design has been reviewed extensively in the past and we consider it to be acceptable.

We have reviewed the capability for testing engineered safety feature circuits during reactor operation. Resistance tests will be used for routine determinations of the operability of the master and slave relay coils. The circuits upstream of these relays can be partially tested during operation. During plant shutdown, circuits can be tested completely by coincident tripping of instrument channels and a consequent operation of the master and slave relays in the entire downstream initiating system. We have concluded that this testing capability is acceptable for Indian Point Unit 2.

8.3 Off-Site Power

Two 138 kilovolt (kV) lines connect the Buchanan switchyard to the Millwood switching station, which in turn is connected to the Consolidated Edison grid and the Niagara Mohawk and Connecticut Light and Power systems. Two additional 138 kV lines, using a separate route from the first two lines, connect the switchyard to the Orange and Rockland tie.

The applicant stated that an analysis of the transmission system has indicated that the system is stable for the loss of any generating unit including Indian Point Unit 2.

A single 138 kV line connects the Buchanan switchyard to Indian Point Unit 2. In addition, three 13 kV lines connect the switchyard to Indian Point Unit 1. Three 138/13 kV transformers in the switchyard feed these three 13 kV lines. While the 138 kV system is the normal supply for the auxiliary load associated with plant engineered safety features, one of the three Indian Point Unit 1 13 kV lines is available to provide power via automatic switching to Indian Point Unit 2 through a 13/6.9 kV transformer. By switching circuit breakers in Indian Point Unit 1, the other two 13 kV lines can also be made available to provide power to Indian Point Unit 2. As the 13/6.9 kV supply is not capable of carrying the total plant auxiliary load for Indian Point Unit 2, the main coolant pumps and the circulating water pumps must be tripped off before the supplies are switched.

We conclude that the off-site power supply provides an adequate source of power for the engineered safety features and safe shutdown loads.

8.4 Onsite Power

Onsite power is supplied by three independent diesel generator sets connected in a separate bus configuration such that there is no automatic closure of tie breakers between the three buses to which the generators are connected. The redundant engineered safety feature (ESF) loads are arranged on the three separate buses such that failure of a single bus will not prevent the required ESF performance under accident conditions. The design engineered safety feature and safe shutdown loads per diesel generator are 1813, 2210, and 2353 HP for the first one-half hour following a loss-of-coolant accident. The loads are then changed to 2438, 2235, and 2043 HP for the recirculation phase of the emergency core cooling system operation. On the basis of our evaluation, we have determined that the appropriate diesel generator ratings are 2200 HP continuous, and 2460 HP for 2,000 hours. We note that some of the estimated emergency loads are above the continuous rating of the machines, but below the 2,000 hour ratings. We consider that this margin is acceptable for Indian Point Unit 2.

Each diesel generator is started automatically upon initiation of emergency core cooling system operation or upon under-voltage on its corresponding 480-volt emergency bus. The generators are housed in a separate Class 1 (seismic) structure. On-site diesel fuel storage capacity provides a minimum of seven days operation at the required safety feature loads. These design and operating features are acceptable for Indian Point Unit 2.

Our review of the ac auxiliary power system has disclosed that there is adequate capacity and an adequate degree of physical and electrical separation of redundant features. The 125 volt dc system consists of two individually housed batteries. The dc system is divided into two buses with a battery and battery charger for each bus. Each of the two station batteries has been sized to carry its expected loads for a period of two hours following a plant trip at a loss of all ac power.

We conclude that the onsite emergency power system is acceptable.

8.5 Cable Installation

We have reviewed the applicant's cable installation relative to the preservation of the independence of redundant channels by means of separation, and relative to the prevention of cable fires through proper cable rating and tray loading. This has been performed by reviewing the cable installation criteria and method of layout design and by field inspection of electrical cable installation during construction.

A single electrical tunnel carries the electrical cables from the electrical penetration area of the containment to the control building. This tunnel carries all of the electrical cables except the power cables for the reactor coolant pumps, the pressurizer heater cables, and the control rod power cables. The cables in the tunnel are arrayed on either side of a three-foot aisle in trays or ladders. Separation is provided for in the form of distance, metal separators, or transite barriers. The electrical tunnel does not contain any spliced cable connections. Therefore, the probability of a fire is reduced. Further, a fire detection system and an automatically operated water spray system are provided in the tunnel. Tunnel cooling is provided for by redundant cooling fans. On the basis of adequate separation within the tunnel, a minimum number of heat producing cables and features, redundant cooling systems, and fire detection and spray systems we conclude that the single electrical tunnel is acceptable.

Sixty electrical penetrations are provided in a single electrical penetration area to provide for entry of signal, control, and power cables into the containment. The penetrations are located on three-foot centers, both horizontally and vertically, and are of the hermetically sealed type. As a result of our review, fire barriers in the form of transite sheets were added to separate the power cable penetration from the instrument and control cable penetrations. In addition, as a result of our review certain modifications were made to the cabling in the penetration area, including shortening of cable runs and elimination of cable loops. The segregation of power cables and the shortening of the cable runs reduces the probability of failure by fire and on this basis, we consider the single electrical penetration area acceptable for Indian Point Unit 2.

The applicant has performed a design audit to verify the separation of redundant engineered safety feature power and control electrical cabling. A design review of instrument cabling was also performed on a sample basis.

On the basis of our review of cable installation at Indian Point Unit 2, we conclude that the resulting cable layout, as installed, is acceptable.

8.6 Environmental Testing

Westinghouse has conducted an environmental test program for the instrumentation and controls that are located inside containment and that must function in the environment following a loss-of-coolant accident. We have reviewed the results of this testing program and conclude that the essential instrumentation and controls will function properly in the accident environment.

9.0 RADIOACTIVE WASTE CONTROL

Liquid and gaseous waste handling facilities are designed to process waste fluids generated by the plant so that discharge of liquid and gaseous effluents to the environment will be minimized. Liquid waste is processed both by direct removal of radioactive material with ion exchange resins and by evaporative separation. Using these methods the volume of radioactive waste will be greatly concentrated and the purified liquid streams will either be reused or discharged. Small quantities of radioactive liquid waste will be released routinely to the condenser circulating water discharge canal common to all three units where the waste will be diluted and discharged to the Hudson River.

The limits on routine radwaste releases from the three units that are planned for operation at the Indian Point site will require that the combined releases from the three units when added together be within the limits specified in 10 CFR Part 20. This requirement is stated in Section 3.9 of the Technical Specifications for both liquid and gaseous effluents.

The liquid effluent releases from the three nuclear facilities will be discharged from a common discharge canal into the Hudson River. The nearest sources of public drinking water supplies from the Hudson River are located at Chelsea, New York (backup water supply for New York City) and at the Castle Point Veterans Hospital, 22 and 20.5 miles upstream of the Indian Point site, respectively.

of the Commission's regulations. For gaseous halogens and particulates with half-lives greater than eight days, the applicable limits of the Technical Specifications are less than 1% of the limits given in 10 CFR Part 20. The Technical Specifications also require that the maximum release rate of gaseous waste not exceed the annual average limit.

Based on our review we conclude that the means provided by the applicant for the disposal of radioactive waste are substantially the same as those we have approved for other facilities and are acceptable. We also conclude that acceptable means are provided and will be used to keep the release of radioactivity from the plant within ranges that we consider to be as low as practicable.

10.0 AUXILIARY SYSTEMS

The auxiliary systems necessary to assure safe plant shutdown include (1) the chemical and volume control system, (2) the residual heat removal system, (3) the component cooling system, and (4) the service water system. The systems necessary to assure adequate cooling for spent fuel include (1) the spent fuel pool cooling system, (2) the fuel handling system, and (3) the service water system. The designs for these systems are substantially the same as those we reviewed and found acceptable for the Ginna plant.

10.1 Chemical and Volume Control System

The chemical and volume control system (1) adjusts the concentration of boric acid for reactivity control, (2) maintains the proper reactor coolant inventory and water quality for corrosion control, and (3) provides the required seal water flow to the reactor coolant pumps. The amount of boric acid to be added to the core for reactivity control is determined by the operator. The addition of unborated water as a result of operator error could result in an unintentional dilution during refueling, reactor startup, and power operation. The applicant's analysis indicated that because of the slow rate of dilution there is ample time for the operator to become aware of the dilution and to take corrective action. The applicant is actively participating in the development of a device for continuous monitoring of the reactor coolant boron concentration and will evaluate the feasibility of installing such a monitor when developed.

Our review of the chemical and volume control system emphasized those portions involved in routine and emergency injection of concentrated boric acid. We conclude that the design is acceptable.

10.2 Auxiliary Cooling Systems

Subsystems for auxiliary cooling are the component cooling system, the residual heat removal loop, the spent fuel pool cooling loop, and the service water system. The piping for these three systems is designed to the ANSI B31.1 Code for Pressure Piping.

These systems are equivalent in purpose and design to those of other recently licensed plants. On the basis of our review of this plant and others using the similar systems, we have concluded that these systems are acceptable.

10.3 Spent Fuel Storage

The fuel handling system is designed to transfer spent fuel to the storage pool and to provide storage for new fuel. The spent fuel storage facility is basically the same in capacity and design as those used in previously licensed pressurized water reactor plants. The fuel pool is sized to accommodate spent fuel from 1-1/3 core loadings.

As in other designs, mechanical stops will be incorporated in the crane to restrict motion of the spent fuel cask to its assigned area, adjacent to one side of the fuel storage pool. In addition, the spent fuel racks in the area adjacent to the fuel cask storage location would be used only in the event that a complete core is unloaded and one-third of a core from a previous unloading is already in storage.

The pool floor is located below grade level and founded on solid rock. Structural damage from a dropped fuel cask would not result in a rapid loss of water from the pool. Makeup water can be supplied from the demineralizer water supply at a flow rate of 150 gpm. Additional water can be provided in an emergency by the use of temporary hookups to other sources.

As a consequence of our evaluation of the potential consequences of a postulated fuel handling accident, the applicant has agreed to provide charcoal filters in the refueling building to reduce the calculated offsite doses that might result in the event of a fuel handling accident in the refueling building. The installation of the filters will be completed during the first year of full power operation.

We conclude that the designs of the spent fuel storage pool and the fuel handling system are acceptable.

11.0 ANALYSES OF RADIOLOGICAL CONSEQUENCES FROM DESIGN BASIS ACCIDENTS

11.1 General

In order to assess the safety margins of the plant design, a number of operating transients were considered by the applicant, including rod withdrawal during startup and at power, moderator dilution, loss of coolant flow, loss of electrical load, and loss of ac power. The reactor control and protection system is designed so that corrective action is taken automatically to cope with any of these transients. Based on our evaluation of the information submitted by the applicant and our evaluations of other PWR designs at the operating license stage, we conclude that the Indian Point Unit No. 2 control and protection system design is such that these transients can be terminated without damage to the core or to the reactor coolant boundary, and with no offsite radiological consequences.

The applicant and we have evaluated the consequences of potential accidents, including a control rod ejection accident, an accident involving rupture of a gas decay tank, a steamline break accident, a steam generator tube rupture accident, a loss-of-coolant accident, and a refueling accident.

The calculated offsite radiological doses that might result from the control rod ejection accident, and the accident involving rupture of a gas decay tank are well within the 10 CFR Part 100 guidelines.

The consequences of the steamline break and the steam generator tube rupture accidents can be controlled by limiting the permissible concentrations of radioactivity in the primary and secondary coolant systems. The Technical Specifications for the Indian Point Unit No. 2 facility limit the primary and secondary coolant activity concentrations such that the potential 2-hour doses at the exclusion radius that we calculate for these accidents do not exceed 1.5 Rem to the thyroid or 0.5 Rem to the whole body.

Our evaluations of the loss-of-coolant accident and the refueling accident are discussed in the following sections.

11.2 Loss-of-Coolant Accident

The design basis loss of coolant accident (LOCA) for the Indian Point Unit No. 2 plant is similar to that evaluated for other PWR plants in that a double-ended break in the largest pipe of the reactor coolant system is assumed.

Although the basis for the design of the emergency core cooling system is to limit fission product release from the fuel, in our conservative calculation of the consequences of the LOCA we have assumed that the accident results in the release of the following percentages of the total core fission product inventory from the core: 100% of the noble gases, 50% of the halogens, and 1% of the solids. In addition, 50% of the halogens that are released from the core is assumed to plate out onto internal surfaces of the containment building or onto internal components and is not available for leakage. We assume that 10% of the iodine available for leakage from the containment is in the form of organic iodide, and that 5% is in the form of particulate iodine. The reactor is assumed to have been operating at a power of 3217 MWt prior to the accident. The primary containment is assumed to leak at a constant rate of 0.1 percent of the containment volume per day for the first day and 0.05 percent per day thereafter. We evaluated the iodine removal capability of the sodium hydroxide containment spray system and assumed an inorganic iodine removal constant of 4.5 per hour for the spray system. We evaluated the iodine removal capability of the iodine impregnated charcoal filter system and assumed a removal constant of 0.49 per hour for inorganic iodine and a removal constant of 0.048 per hour for organic iodine. Iodine particulates are assumed to be removed by the high efficiency particulate air filters. The inhalation rate of a person offsite is assumed to be 3.5×10^{-4} cubic meters per second.

For the calculation of the two-hour dose at the site boundary we used an atmospheric dispersion factor corresponding to Pasquill Type "F" stability, with a 1 meter per second wind speed and an appropriate building wake effect. We calculated the potential doses at the site boundary for this 2 hour period to be 180 Rem to the thyroid and 4 Rem to the whole body. At the low population zone boundary our calculated potential doses for a 30-day period are 270 Rem to the thyroid and 7 Rem to the whole body.

In evaluating the above doses, no credit was given for the isolation valve seal water injection system, the penetration pressurization system, or the weld channel pressurization system. Operation of these systems, which interpose a high gas pressure or seal water area between the containment and the outside atmosphere at all points where leakage might occur, should significantly reduce the leakage rate from the containment, and, thus reduce the doses following an accident. These systems are well designed and tested, and should be available in the event of an accident (see Section 7.3). We did not consider the effect of these systems in our dose calculations because it is inherently difficult to accurately measure leakage rates of less than 0.1% per day by current testing methods.

The control room for Indian Point Unit No. 2 was not designed to meet the requirements we have imposed in more recent construction permit reviews, that the dose for the course of the accident to occupants of the control room be limited to 5 Rem to the whole body and 30 Rem to the thyroid. In order to provide additional protection to the control room occupants in the event of a loss-of-coolant accident, the applicant has equipped the control room with protective clothing and self-contained air respirators for the operators. In view of these provisions, we have concluded that the control room, as constructed, is acceptable in this regard.

11.3 Fuel Handling Accident

We have evaluated the potential consequences of a fuel handling accident, in which it is postulated that a fuel assembly is dropped in the spent fuel pool or transfer canal. We assumed that: (1) all 204 rods in the dropped bundle are damaged, (2) the accident occurs 90 hours after shutdown of the core from which the dropped bundle has been removed, (3) 20% of the noble gases and 10% of the iodine in the dropped fuel bundle are released to the refueling water and the dropped fuel bundle has been removed from a region of the core which has been generating 1.43 times the average core power, (4) 90% of the released iodine is retained in the refueling water, (5) the fission products released from the pool are discharged to the atmosphere by the building recirculation system through charcoal filters with an iodine removal efficiency of 90%, and (6) the same meteorological conditions exist as were assumed for the loss-of-coolant accident. The resultant calculated doses at the site boundary are 146 Rem to the thyroid and less than 4 Rem to the whole body.

11.4 Conclusions

We have calculated offsite doses for the design basis accidents that have the greatest potential for offsite consequences using assumptions consistent with those we have used in previous safety reviews of PWR plants and have found the resulting calculated doses to be less than the guideline values of 10 CFR Part 100.

(3) review and report upon all proposed changes to the Technical Specifications; (4) conduct unannounced spot inspections of plant monitoring operations; (5) review and report upon any activity, the occurrence or lack of which may affect the safe operation of the nuclear plant; and (6) convene, at the request of the nuclear power generation manager or a nuclear plant general superintendent or chairman or vice chairman of the committee, to review and act upon any matter they may deem necessary.

Westinghouse will participate in the startup and initial operation of the plant and will continue to make available technical support to the Indian Point Unit 2 staff during operation of the facility.

We conclude that the applicant's organization is acceptably staffed and technically qualified to perform its operational duties subject to satisfactory completion of licensing examinations of personnel requiring licenses.

12.3 Emergency Planning

The site emergency plan for the Indian Point site describes the emergency organization and its responsibilities. The scope of the emergency plan includes consideration of local contingencies, site contingencies, general (off-site) contingencies, implementation levels for each contingency, notification channels, the support provided by civil authorities, protective measures for each contingency, communications facilities, and training drills. The applicant has provided an extensive description of the medical support that will be available although it is not incorporated explicitly in the plan. The planned medical support provides for emergency treatment of plant personnel both at the site and at a designated hospital where facilities equipment and medical personnel to handle radiation contaminated injured personnel will be available.

We conclude that the applicant's emergency plan is acceptable for Indian Point Unit 2.

12.4 Industrial Security

The immediate plant area (restricted area), including Indian Point Unit 1 will be enclosed by a fence. Access to the restricted area for all personnel will be through manned gatehouses or locked gates which are under the direct control of the station security forces. Security guards will make routine patrols of all property within the site boundary and outside the restricted area and are required to make hourly reports to the central control room.

The controlled area of Indian Point Unit 2 will include the containment, the fuel storage building, the primary auxiliary building, and the emergency diesel generator building. Normal access to these areas is through the existing security room for Indian Point Unit 1. All other doors and hatches leading into the controlled area will be locked and will be supervised by means of door switches connected to the open door alarm board in the security room, and the category alarm board in the Indian Point Unit 1 central control room. The containment personnel hatch doors have remote indicating lights and annunciators that are located in the control room and that indicate the door operational status.

Offsite applicant employees must identify themselves at the main gate prior to admission to the restricted area, receive approval for entry by the general superintendent or his designated representative, and sign in on an admission sheet. If access into the controlled area is approved, they must be accompanied by a qualified guide.

We conclude that the applicant has taken reasonable measures to provide for the security of the facility.

13.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in an operating license define safety limits and limiting safety system settings, limiting conditions for operation, periodic surveillance requirements, certain design features, and administrative controls for the operating plant. These specifications cannot be changed without prior approval of the AEC. The applicant's initial proposed Technical Specifications, presented in Amendment No. 20, have been modified as a result of our review to describe more definitively the allowable conditions for plant operation. The Technical Specifications as approved by the regulatory staff, may be examined in the Commission's Public Document Room.

Based upon our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits and that means are provided for keeping the release of radioactivity from the plant within ranges that we consider as low as practicable. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features to mitigate the consequences of unlikely accidents will be available.

14.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The ACRS reported on the application for construction of the Indian Point Unit 2 at the proposed site in a letter dated August 16, 1966. The applicant has been responsive to the recommendations made by the ACRS in that letter, and we conclude that the matters raised have been resolved satisfactorily during the design and construction of the Indian Point Unit 2.

The ACRS reported on its review of the application for an operating license for Indian Point Unit 2 in their letter, dated September 23, 1970, attached as Appendix B.

In its letter, the ACRS made several recommendations and noted several items all of which have been considered in the indicated sections of our evaluation. These include: (1) reevaluation of potential flooding at the Indian Point site (Section 3.4), (2) additional seismic reinforcing at the Indian Point Unit No. 1 superheater building and truncation of the superheater stack (Section 6.2), (3) reactor design, power distribution, and control of potential xenon oscillations (Section 4.2), (4) containment design and isolation (Sections 6.2 and 7.3), (5) containment cooling and iodine removal systems (Section 7.2), (6) emergency core cooling system and removal of the reactor pit crucible (Section 7.1), (7) post-accident hydrogen control (Section 7.4),

(8) charcoal filters in the refueling building (Section 10.3), (9) reactor core instrumentation (Section 4.2), (10) reactor protection with only three of four loops in service (Section 8.1), (11) inservice vibration monitoring and loose parts detection (Section 5.9), (12) fuel failure detection (Section 5.9), (13) availability requirements for primary coolant leak detection systems (Section 5.7), (14) pressure vessel fracture toughness (Section 5.2), (15) integrity of high burnup fuel during design transients (Section 4.3), and (16) common mode failure and anticipated transients without reactor scram (Section 8.1).

The ACRS concluded in its letter that if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

15.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and all of the directors and principal officers of the applicant are United States citizens.

The applicant is not owned, dominated or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes, is involved. For these reasons and in the absence of any information to the contrary, we have found that the activity to be performed will not be inimical to the common defense and security.

16.0 FINANCIAL QUALIFICATIONS

The Commission's regulations that relate to the financial data and information required to establish financial qualifications for an applicant for an operating license are 10 CFR Part 50.33(f) and 10 CFR Part 50 Appendix C. The Consolidated Edison Company's application as amended by Amendment No. 21 thereto, and the accompanying certified annual financial statements provided the financial information required by the Commission's regulations.

These submittals contain the estimated operating cost for each of the first five years of operation plus the estimated cost of permanent shutdown and maintenance of the facility in a safe condition. The estimated operating costs are $10.0 million for 1971 (the first year of operation), $14.8 million for 1972, $12 million for 1973, $10.9 million for 1974 and $10.7 million for 1975 (Amendment No. 21). Such costs include the costs of operating and maintenance and fuel. The applicant's estimate of the cost of permanently shutting down the facility and maintaining it in a safe condition is (1) $265,000 for the first year of shutdown and $50,000 for each year thereafter if the reactor core is removed from the vessel, and (2) $240,000 per year if the core is not removed.

We have examined the certified financial statements of the Consolidated Edison Company to determine whether the Company is financially qualified to meet these estimated costs. The information contained in the 1969 financial report indicates that operating revenues for 1969 totaled $1,028.3 million; operating expenses (including taxes) was $830.5 million; the interest on the long-term debt was earned 2.3 times; and the net income for the year was $127.2 million, of which $102.1 million was distributed as dividends to the stockholders, and the remainder of $25.1 million was retained for use in the business. As of December 31, 1969, Company's assets totaled $4,069.6 million, most of which was invested in utility plant ($3,793.3 million), and earnings reinvested in the business were $426.1 million. Financial ratios computed from the 1969 statements indicate a sound financial condition, (e.g., long-term debt to total capitalization--0.52, and to net utility plant--0.52; net plant to capitalization--0.994; the operating ratio--0.81; and the rates of return on common--7.7%; on stockholder's investment--6.9%; and on total investment--4.9%). The record of the Company's operations over the past 5 years reflects that operating revenues increased from $840 million in 1965 to $1,028 million in 1969; net income increased from $111.8 million to $127. million; and net investment in utility plant from $3,170 million to $3,793 million. Moody's Investors Service (August 1969 edition) rates the Company's first mortgage bonds as A (high-medium grade). The Company's current Dun and Bradstreet rating (July 1970) is AaA1.

Our evaluation of the financial data submitted by the applicant, summarized above, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR Part 50.33(f) with respect to the operation of Indian Point Unit 2. A copy of the staff's financial analysis is attached as Appendix H.

17.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licensees for facilities such as power reactors under 10 CFR Part 50.

17.1 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The Consolidated Edison Company, is with respect to Indian Point Unit 2, subject to the foregoing requirements, and has taken the following steps with respect thereto.

The Company has furnished to the Commission proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association policy (Nuclear Energy Liability Policy, facility form) Nos. NF-100.

Further, the Company executed Indemnity Agreement No. B-19 with the Commission as of January 12, 1962, which was amended to cover its pertinent preoperational fuel storage under license SNM-1108 on March 4, 1969. The Company has paid the annual indemnity fee applicable to preoperational fuel storage.

17.2 Operating License

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from, preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for reactors which have a rated capacity of 100,000 electrical kilowatts or more is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently $82 million.

Accordingly, no license authorizing operation of Indian Point Unit 2 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended and that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operation. The amount of financial protection required for a reactor having the rated capacity of this facility would be $82 million. Consolidated Edison Company will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of $30 per each thousand kilowatts of thermal capacity authorized in its operating license.

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licensees, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

18.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that:

1. The application for facility license filed by the Consolidated Edison Company of New York, Inc., dated December 6, 1965, as amended (Amendments Nos. 9 through 25, dated October 15, 1968, October 13, 1969, October 24, 1969, November 21, 1969, December 29, 1969, January 27, 1970, March 2, 1970, March 30, 1970, April 17, 1970, June 3, 1970, July 14, 1970, July 17, 1970, July 28, 1970, July 29, 1970, August 13, 1970, August 28, 1970, and November 12, 1970, respectively) complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and
2. Construction of the Indian Point Nuclear Generating Unit No. 2 (the facility) has proceeded and there is reasonable assurance that it will be completed, in conformity with Provisional Construction Permit No. CPPR-21, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
3. The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

4. There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
5. The applicant is technically and financially qualified to engage in the activities authorized by this operating license, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
6. The applicable provisions of 10 CFR Part 140 have been satisfied; and
7. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

Prior to any public hearing on the matter of the issuance of an operating license to Consolidated Edison for Indian Point Unit No. 2, the Commission's Division of Compliance will prepare and submit a supplement to this Safety Evaluation which will deal with those matters relating to the status of construction completion and conformaty of this construction to the provisional construction permit and the application. Before an operating license will be issued to Consolidated Edison for Indian Point Unit No. 2, assuming such a license is authorized following the public hearing, the facility must be completed in conformity with the provisional construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Division of Compliance prior to license issuance.

APPENDIX A

CHRONOLOGY OF

REGULATORY REVIEW OF THE CONSOLIDATED EDISON COMPANY

INDIAN POINT NUCLEAR GENERATING PLANT UNIT NO. 2

(SUBSEQUENT TO CONSTRUCTION PERMIT NO. CPPR-21

ISSUED ON OCTOBER 14, 1966)

1.	April 17, 1967	Submittal of Amendment No. 6 containing design information on the Emergency Core Cooling System and other areas as requested by the ACRS in their letter to the Chairman AEC, of 8/16/66.
2.	July 18, 1967	Meeting with applicant to discuss revised design of Emergency Core Cooling System and other areas as per Amendment No. 6.
3.	August 2, 1967	Letter to applicant requesting additional information on subjects addressed by the ACRS in their letter of 8/16/66.
4.	October 16, 1967	Submittal of Amendment No. 7 in response to DRL request of August 2, 1967.
5.	October 31, 1967	Submittal of Amendment No. 8, revised pages for Amendment No. 7.
6.	December 28, 1967	ACRS Subcommittee meeting to discuss emergency core cooling system, reactor pit crucible, primary coolant system, other areas.
7.	January 30, 1968	Submittal of "Report on the Containment Building Liner Plate Buckle in the Vicinity of the Fuel Transfer Canal".
8.	February 2, 1968	Meeting with applicant to discuss content of Amendments No. 6, 7, and 8.
9.	February 13, 1968	Meeting with applicant to complete discussion of February 2, 1968.

10.	March 8, 1968	ACRS Full Committee meeting to discuss Emergency Core Cooling System; reactor internals; primary coolant system, design, fabrication, in-service inspection, and leak detection; core design; reactor pit crucible; and containment liner quality control and stress analysis.
11.	October 15, 1968	Consolidated Edison Company filed application for an Operating License for the IP-2 Plant. Amendment 9, Volumes 1, 2, 3, & 4.
12.	March 5, 1969	AEC-DRL requested additional information on medical and emergency plans.
13.	March 12, 1969	AEC-DRL staff met with Con Ed personnel to discuss scheduling of regulatory review of application for operating license.
14.	April 3, 1969	AEC-DRL staff met with Con Ed personnel to discuss structural and seismic design and tornado protection.
15.	April 16, 1969	AEC-DRL staff met with Con Ed to discuss accidental and normal radioactivity release from the IP-2 plant.
16.	April 28, 1969	Con Ed requested extension of completion date for construction of the IP-2 plant.
17.	May 2, 1969	AEC-DRL staff and Nathan M. Newmark, seismic design consultant, met with Con Ed personnel at the IP-2 site to discuss seismic design and review status of construction and site inspection.
18.	May 19, 1968	AEC-DRL staff issued an order extending completion date for construction of the IP-2 plant to June 1, 1970.

19.	August 4, 1969	Request to applicant for additional information on site and environment, reactor coolant system, containment system, engineered safety features, instrumentation and control, electrical systems, waste disposal and radiation protection, conduct of operations, and accident analysis.
20.	August 22, 1969	AEC-DRL staff requests copies of monitoring reports and status of actions on Fish and Wildlife recommendations.
21.	August 23, 1969	ACRS Subcommittee meeting on tornado protection, emergency planning, permanent in-core instrumentation, adequacy of onsite emergency power, and containment isolation.
22.	September 24, 1969	Meeting with applicant to discuss Westinghouse presentation on power distribution detection and control in Indian Point 2.
23.	October 13, 1969	Submittal of Amendment 10 (Supplement #1) responses to AEC regulatory staff's request of March 5, 1969, on medical plans and partial answers to AEC regulatory staff's request for additional information of August 4, 1969.
24.	October 24, 1969	Submittal of Amendment No. 11, replacement pages and responses to AEC regulatory staff's request for additional information of August 4, 1969, on Sections 1, 4, 5, 6, 7, 12, and 14 of the FSAR.
25.	November 13, 1969	Request for additional information on reactor, reactor coolant system, containment system, engineered safety features, auxiliary and emergency systems, initial tests and operations, and accident analysis.
26.	November 21, 1969	Submittal of Amendment No. 12, additional and replacement pages to be inserted into the FFDSAR and further responses to AEC regulatory staff's request for additional information of 8/4/69 on Sections 1, 4, 7, 8 and 11 of the FFDSAR.

27.	December 10, 1969	Meeting with applicant to review electrical drawings including AC power, DC power, Reactor Protection System, and Engineered Safety Features.
28.	December 30, 1969	Meeting with applicant and Westinghouse Electric Corporation to continue detailed review of electrical drawings including Reactor Protection System and Engineered Safety Features.
29.	January 16, 1970	Meeting with applicant to review and discuss electrical drawings including Reactor Protection System and Engineered Safety Features.
30.	January 21, 1970	Meeting with applicant & Westinghouse Electrical Corporation on technical specifications.
31.	January 27, 1970	Submittal of Amendment No. 14, replacement pages for FSAR & further responses to AEC-DRL questions of 8/4/69 & 11/13/69, chapters 1, 4, 6, 11, 12 & 14.
32.	February 17, 1970	Meeting with applicant for presentation of results of Con Ed's Analysis concerning potential damage to Indian Point 2 and IP-3 from a failure of the IP-1 superheater stack.
33.	March 2, 1970	Submittal of Amendment No. 15, responses to AEC regulatory staff's requests for additional information of 8/4 and 11/13, 1969 and Containment Design Report.
34.	March 10, 1970	Request to applicant for additional financial data.
35.	March 13, 1970	Meeting with applicant to discuss questions concerning core heat transfer and burnout limits, fuel element performance and ECCS performance during a LOCA.

36.	March 19, 1970	Meeting with applicant, Westinghouse presentation on iodine removal system for IP-2.
37.	March 26, 1970	Meeting with applicant to discuss analysis of fresh water flood and changes to electrical systems.
38.	March 30, 1970	Submittal of Amendment No. 16, additional and replacement pages for the FSAR and further responses to the AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
39.	April 25, 1970	ACRS Subcommittee meeting and meeting with applicant on instrumentation and control, and anticipated transients with failure to scram.
40.	April 17, 1970	Submittal of Amendment No. 17, additional and replacement pages to be inserted into the FSAR and further responses to AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
41.	April 29, 1970	Meeting with applicant to discuss seismic and structural design questions for IP-2.
42.	May 5, 1970	Meeting with applicant to discuss failure mode analysis of the engineered safety feature manual actuation panel.
43.	May 11, 1970	ACRS Subcommittee meeting at the Indian Point 2 site to discuss instrumentation and control and Electrical Systems.
44.	May 12, 1970	AEC issued Order extending completion date for construction of the IP-2 plant to June 1, 1971.
45.	May 28, 1970	ACRS Subcommittee meeting to discuss loss-of-coolant accident, anticipated transients with failure to scram.
46.	June 3, 1970	Submittal of Amendment No. 18, additional and revised pages for the FSAR in response to AEC regulatory staff request for additional information.

47.	June 11, 1970	ACRS full Committee meeting to consider design of engineered safety feature manual actuation panel and operation with less than four loops.
48.	June 17, 1970	Meeting with applicant to discuss consequences of turbine missiles, sensitized stainless steel control room accident dose, hydrogen recombiner.
49.	July 15, 1970	Submittal of Amendment No. 19 (Supplement 10), additional and revised pages for the FSAR and Flooding Evaluation report.
50.	July 20, 1970	Submittal of Amendment No. 20, (Supplement 11) proposed Technical Specifications.
51.	July 24, 1970	Request for additional information on emergency core cooling, reactor coolant system, instrumentation and control, electrical systems, conduct of operations and accident analysis.
52.	July 28, 1970	Submittal of Amendment No. 21, Con Ed Annual Report.
53.	July 28 and 29, 1970	ACRS Subcommittee meeting to discuss technical specifications, flood protection, Unit No. 1 superheater stack failure and containment sprays.
54.	July 30, 1970	Submittal of Amendment No. 22, (Supplement 12), revised pages for FSAR in response to request for additional information.
55.	August 7, 1970	Meeting with applicant to discuss technical specifications.
56.	August 13, 1970	ACRS full Committee meeting to discuss the matters addressed in our July 2, 1970 report.
57.	August 14, 1970	Submittal of Amendment No. 23 (Supplement 13), answers to request for additional information issued July 24.

58.	August 18, 1970	Meeting to discuss licensed operator requirements.
59.	August 28, 1970	Submittal of Amendment No. 24 (Supplement 14). Revised pages to the FSAR.
60.	September 1, 1970	Meeting with applicant regarding performance of Emergency Core Cooling System.
61	September 9, 1970	Meeting with the applicant to discuss Technical Specifications.
62.	October 21, 1970	Request to applicant for a report on analysis of laminations in base plate material of the IP-2 pressurizer.
63.	October 29, 1970	Meeting with applicant to review technical specifications for the Indian Point 2 plant.
64.	November 1970	Submittal of Amendment 25 (Supplement 15), changes to technical specifications and to FSAR.

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

SEP 23 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by Consolidated Edison Company of New York, Inc., for authorization to operate the Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the Committee's 95th, 98th, 122nd, and 124th meetings, and at Subcommittee meetings on August 23, 1969, March 13, 1970, April 25, 1970, May 28, 1970, July 28-29, 1970, and September 15, 1970. Subcommittees also met at the site on December 28, 1967 and May 11, 1970. The Committee last reported on this project to you on August 16, 1966. During the review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants, and with representatives of the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Indian Point site is located in Westchester County, New York, approximately 24 miles north of the New York City limits. The minimum radius of the exclusion area for Unit No. 2 is 520 meters and Peekskill, the nearest population center, is approximately one-half mile from the unit. Also at this site are Indian Point Unit 1, which is licensed for operation at 615 MWt, and Unit 3, which is under construction.

The applicant has re-evaluated flooding that could occur at the site in the event of the probable maximum hurricane and flood, in the light of more recent information, and has concluded that adequate protection exists for vital components and services.

Additional seismic reinforcement being provided for the Indian Point Unit No. 1 superheater building and removal of the top 80 ft. of the superheater stack will enable the stack to withstand winds in the range of 300-360 mph corresponding to current tornado design criteria. Since the reinforcement of the superheater building, which supports the stack, enables the stack to resist wind loads of a magnitude most likely to be experienced from a tornado, the Committee believes that removal of the top 80 ft. of the stack, to enable it to resist the maximum effects from a tornado, may be deferred until a convenient time during the next few years, but prior to the commencement of operation of Indian Point Unit No. 3. The applicant has stated that truncation of the stack will have no significant adverse effect on the environment.

The Indian Point Unit No. 2 is the first of the large, four-loop Westinghouse pressurized water reactors to go into operation, and the proposed power level of 2758 MWt will be the largest of any power reactor licensed to date. The nuclear design of Indian Point Unit No. 2 is similar to that of H. B. Robinson with the exception that the initial fuel rods to be used in Indian Point Unit No. 2 will not be prepressurized. Part-length control rods will be used to shape the axial power distribution and to suppress axial xenon oscillations. The reactor is designed to have a zero or negative moderator coefficient of reactivity, and the applicant plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses related to these tests.

Unit 2 has a reinforced concrete containment with an internal steel liner which is provided with facilities for continuous pressurization of weld and penetration areas for leak detection, and a seal-water system to back up piping isolation valves. In the unlikely event of an accident, cooling of the containment is provided by both a containment spray system and an air-recirculation system with fan coolers. Sodium hydroxide additive is used in the containment spray system to remove elemental iodine from the post-accident containment atmosphere. An impregnated charcoal filter is provided to remove organic iodine.

Major changes have been made in the design of the emergency core cooling system as originally proposed at the time of the construction permit review. Four accumulators are provided to accomplish rapid reflooding of the core in the unlikely event of a large pipe break, and redundant pumps are included to maintain long-term core cooling. The applicant has analyzed the efficacy of the emergency core cooling system and concludes that the system will keep the core intact and the peak clad temperature well below the point where zircaloy-water reaction might have an adverse effect on clad ductility and, hence, on the continued structural integrity of the fuel elements. The Committee believes that there is reasonable assurance that the Indian Point Unit No. 2 emergency core cooling system will perform adequately at the proposed power level.

The Committee concurs with the applicant that the reactor pit crucible, proposed at the time of the construction permit review, is not essential as a safety feature for Indian Point Unit No. 2 and need not be included.

To control the concentration of hydrogen which could build up in the containment following a postulated loss-of-coolant accident, the applicant has provided redundant flame recombiner units within the containment, built to engineered safety feature standards. Provisions are also included for adequate mixing of the atmosphere and for sampling purposes. The capability exists also to attach additional equipment so as to permit controlled purging of the containment atmosphere with iodine filtration. The Committee believes that such equipment should be designed and provided in a manner satisfactory to the Regulatory Staff during the first two years of operation at power.

The applicant plans to install a charcoal filter system in the refueling building to reduce the potential release of radioactivity in the event of damage to an irradiated fuel assembly during fuel handling. This installation will be completed by the end of the first year of full power operation.

The reactor instrumentation includes out-of-core detectors, fuel assembly exit thermocouples, and movable in-core flux monitors. Power distribution measurements will also ordinarily be available from fixed in-core detectors.

The applicant has proposed that a limited number of manual resets of trip points, made deliberately in accordance with explicit procedures, by approved personnel, independently monitored, and with settings to be calibrated and tested, should provide an acceptable basis for the occasional operation of Indian Point Unit No. 2 with only three of the four reactor loops in service. The Committee concurs in this position.

The applicant stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefulness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

The reactor includes a delayed neutron monitor in one hot leg of the reactor coolant system to detect fuel element failure. Suitable operability requirements will be maintained on the several sensitive means of primary system leak detection.

A conservative method of defining pressure vessel fracture toughness should be employed that is satisfactory to the Regulatory Staff.

The applicant stated that existing experimental results and analyses provide considerable assurance that high burnup fuel of the design employed will be able to undergo anticipated transients and power perturbations without a loss of clad integrity. He also described additional experiments and analyses to be performed in the reasonably near future which should provide further assurance in this regard.

The Committee has, in recent reports on other reactors, discussed the need for studies on further means of preventing common failure modes from negating scram action, and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has provided the results of analyses which he believes indicate that the consequences of such transients are tolerable with the existing Indian Point Unit No. 2 design at the proposed power level. Although further study is required of this general question, the Committee believes it acceptable for the Indian Point Unit No. 2 reactor to operate at the proposed power level while final resolution of this matter is made on a reasonable time scale in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept advised.

Other matters relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS letters should, as in the case of other reactors recently reviewed, be dealt with appropriately by the Staff and the applicant in the Indian Point Unit No. 2 as suitable approaches are developed.

The ACRS believes that, if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit No. 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Original Signed by
Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References attached.

References - Indian Point Nuclear Generating Unit No. 2

1. Amendment No. 9 to Application of Consolidated Edison Company of New York for Indian Point Nuclear Generating Unit No. 2, consisting of Volumes I - IV, Final Safety Analysis Report, received October 16, 1968

2. Amendments 10 - 20 to the License Application
3. Amendments 22 - 24 to the License Application

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis Report
Volumes I, II, III and IV dated October 15, 1968

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
November 29, 1968

As pointed out in our comments of October 29, 1965 on Unit No. 2, a primary influence on the meteorological statistics of the Indian Point site seems to be its location in a river valley about a mile wide with terrain rising 600 to 1000 feet on either side. Consequently, wind directions follow a pronounced diurnal cycle with daytime, unstable (lapse) flow in the upriver direction and nighttime, stable flow in the downriver directions. The report documents a 42.4 percent inversion frequency, but it should also be pointed out that inversion conditions are largely confined to the nighttime, downriver flow lasting about 12 hours before changing to lapse or upriver flow. Figure 2.6-1, although in terms of average vectors, shows the marked wind reversals at sunset and sunrise and the rather persistent, channeled flow that can occur during the middle of the night (see the mean direction between 0200 and 0800 hours). The mean wind speeds during this persistent period is about 2.5 m/sec which indicates that 50 percent of the time inversion wind speeds could be less than 2.5 m/sec.

In the absence of specific, joint-frequency wind speed and direction persistence data from the site, a reasonably conservative meteorological model would be to assume for a ground release a 1 m/sec wind speed under inversion conditions in a persistent downriver direction for a period of 8 hours. Taking into account the likelihood of a diurnal wind reversal, a very conservative assumption would be to allow the plume centerline to meander over a 22-1/2° arc under the same conditions for the remainder of the 24-hour period. Again, with no specific on-site wind persistence data, the conservative assumption has been made.

The amount of additional atmospheric diffusion because of the building turbulence can be assessed by the virtual point source expression $[(x + x_o)/x]^{1.5}$ as used by the applicant, which for a value of x_o = 430 m amounts to a factor of 2.5 at the site boundary (520 m) and 1.6 at the low population boundary (1100 m). These values are in close agreement with the method of using a shape factor of 1/2 and a building cross-section of 2000 m^2.

In summary, from data presently available, it would seem reasonably conservative to assume a persistent wind direction for an 8-hour period under inversion conditions and a 1 m/sec wind speed. With the added assumption of a building wake shape factor of 1/2 and a cross-sectional area of 2000 m^2, the resulting 0-8 hr relative concentration would be 6.6×10^{-4} sec m3 at the site boundary and 3.7×10^{-4} at the low population boundary. From Table 14.3.5-3 one can calculate that the applicant's model for the 0-8 hr period results in an average relative concentration of 4.8×10^{-4} and 2.4 sec m^{-3} at the site and low population boundary, respectively.

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis
Amendment No. 12 dated November 21, 1969, and
Amendment No. 14 dated January 27, 1970

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
February 17, 1970

The original documentation of the Indian Point site during the period 1955-1957 indicates that at the 100-ft. height the annual prevailing wind direction is from the north northeast and that in the sector from 22.5 to 42.5 degrees the frequency of inversion, neutral and lapse conditions was 6, 2, and 1 percent, respectively. Within this sector, the shortest site boundary is approximately in a direct line through Units 2 and 3 at a distance of 610 and 380 m, respectively, as measured from figure 2.2-2. It is about 500 m from the Unit 1 stack to this common boundary point. The nearest site boundary, regardless of sector, is where the property line intersects the downriver edge of the site. Although this point is at a distance of 580 m from Unit 2, it is not in the most prevalent wind direction by a considerable amount.

To compute the average annual dilution factor we have assumed the frequencies listed above, averaged over a 20-degree sector with a wind speed of 2, 4 and 3 m/sec, respectively, for inversion (Type F), neutral (Type D), and lapse (Type B) conditions. Assuming no building wake effect our results show the applicant's values for Units 1 and 2 to be reasonably conservative. In the case of Unit 3 we compute an average annual dilution factor of 2.9×10^{-5} sec m^{-3} as compared to the applicant's value of 1.6×10^{-5} sec m^{-3}. The only explanation we have for the ESSA value being twice as high is the use of the building wake effect in the applicant's assumptions.

It is our view that the use of the building wake effect in the long-term average diffusion equation, as was done by the applicant, is inappropriate. It does not seem logical that for the same atmospheric conditions the Sutton equation on page Q 11.10-1 for the long-term model gives <u>more</u> credit for building wake effect than the equivalent short-term model on p. Q 11.10-2. For example at $x = 400$ m assuming $x_0 = 400$ m and $n = 0.5$, the building wake effect, $[(x+x_0)/x_0]^{2-n/2}$, for the long-term equation is 3.4 whereas for the effect in the short-term equation, $[(x+x_0)/x_0]^{2-n}$, the value is 2.8. It is the larger exponent in the former that makes the difference. Also, the fact that one averages in the horizontal dimension over a sector essentially would nullify any added dilution in that dimension because of wake effect.

DEPARTMENT OF THE ARMY
COASTAL ENGINEERING RESEARCH CENTER
5201 LITTLE FALLS ROAD, N.W.
WASHINGTON, D.C. 20016

CERER

21 November 1969

Mr. Roger S. Boyd
Asst. Director for Reactor Projects
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Boyd:

Reference is made to your letters regarding Docket Nos. 50-247, 50-286, 50-342, and 50-343, Consolidated Edison Company of New York's proposed Indian Point Nuclear Generating Units No. 2 and No. 3, and Units No. 4 and No. 5 which are contiguous to Indian Point plant site.

Pursuant with our arrangements, Mr. R. A. Jachowski and Mr. B. R. Bodine of CERC have reviewed all pertinent information contained in the reports from the standpoint of establishment of a design water level. This included the review of the storm surge associated with the Probable Maximum Hurricane (PMH) and wind wave analysis.

We concur with the applicant's finding that the design water level should be 14.5 feet above the mean sea level datum for Units, Nos. 2, 3, 4 and 5. Although this value is acceptable, there are compensating errors in routing procedure employed.

If you have any further questions regarding this matter please let us know.

Sincerely yours,

Edward Willis
EDWARD M. WILLIS
Lieutenant Colonel, CE
Director

APPENDIX E

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

SEP 16 1970

Mr. Harold Price
Director of Regulation
U.S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by R. C. DeYoung is a review of the flood information presented in Amendment No. 19 to the Final Safety Analysis Report for Unit No. 2 Indian Point Nuclear Generating Station. It is presumed that the flood levels for all 3 units at the Indian Point Stations will be based on this amendment. Copies of our earlier reviews, for Unit No. 2 (Aug. 15, 1966) prepared by E. L. Meyer, and for Unit No. 3 (January 6, 1969) prepared by P. J. Carpenter, are attached.

This review was prepared by P. J. Carpenter and has been discussed with members of your staff. We have no objection to your making this review a part of the public record.

Sincerely yours,

Acting Director

Enclosures

Consolidated Edison Company of New York Inc.
Indian Point Nuclear Generating Station Unit No. 2
Docket No. 50-247

The probable maximum flood as defined by the U.S. Army Corps of Engineers, at the site, has been calculated as 1,[illegible],000 cubic feet per second. This discharge is approximately three times greater than the maximum observed flood at Green Island, and is approximately twice the maximum discharge observed for nearby like-sized drainage basins which appear to exhibit similar runoff characteristics. The stage for the maximum probable flood at the site, computed using standard step-backwater procedures, is given as varying between 13.4 and 14.0 ft msl (mean sea level) depending on concurrent tide levels at the Battery. It is shown that none of the dams on the Hudson River and its tributaries would fail during the probable maximum flood. The above results were obtained using conservative assumptions and appear to be reasonable.

The analyses show that the occurrence of the probable maximum flood on Esopus Creek would cause failure of Ashokan Dam some 75 miles upstream of the site. To establish a flood design level at Indian Point various combinations of the following factors were considered: 1) the flow resulting from the Ashokan Dam failure, 2) various concurrent Hudson River flood flows, and 3) various concurrent tide levels at the Battery. The results of these combinations of factors were compared with the stage of the probable maximum flood (14.0 ft msl) and the stage resulting from the probable maximum hurricane plus spring high tide (14.5 ft msl). The most critical combination investigated consisted of the flows from the Ashokan Dam failure caused by the probable maximum flood on Esopus Creek, the concurrent standard project flow (one half the probable maximum flood), the concurrent stage at the Battery corresponding to the standard project hurricane tide level and wind waves of one foot at the site. This stage is given as 15.0 ft msl. The lowest floor elevation of Unit No. 2 is given as 15.25 ft msl.

Other combinations of the above-mentioned factors, such as Ashokan Dam failure and the standard project hurricane or floods larger than the standard project flood on the Hudson River, could produce higher stages at the site. Depending on the degree of conservatism desired, any of these higher stages could also be selected as the design flood level. However, the stage for the combination selected for the design flood level exceeds those given for the probable maximum flood or probable maximum hurricane when these are considered as independent events.

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

APPENDIX F

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Consolidated Edison Company of New York, Inc.

Docket No. 50-247

by

N. M. Newmark
and
W. J Hall

Urbana, Illinois

20 August 1970

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

INTRODUCTION

This report is concerned with the structural adequacy of the containment structures, piping, equipment and other critical components for the Indian Point Nuclear Generating Unit No. 2 for which application for a construction permit and an operating license has been made to the United States Atomic Energy Commission by the Consolidated Edison Company of New York, Inc. The facility is located on the east bank of the Hudson River at Indian Point, village of Buchanan, in upper Westchester County, New York. The site is about 24 miles N of the New York City boundary and 2.5 miles SW of Peeksill, New York.

This report is based on a review of the Final Facility Description and Safety Analysis Report (Ref. 1) and the containment design report (Ref. 2). The report also is based in part on the discussion and inspection resulting from the visit to the site on 2 May 1969 by N. M. Newmark and W. J. Hall in conjunction with Mr. K. Kniel and Mr. M. McCoy of AEC-DRL. A number of topics were discussed with the applicant and his consultants at the time of this visit, and subsequently additional information has become available through supplements to the FSAR and through discussions with the personnel of DRS, DRL, and the applicant and his consultants. A discussion of the adequacy of the structural criteria presented in the Preliminary Safety Analysis Report is contained in our report of August 1966 (Ref. 3), and unless otherwise noted no comment will be made in this report concerning points covered there.

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The design criteria for the containment system and Class I components for this plant called for a design to withstand a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration coupled with other appropriate loadings to provide for containment and safe shut down. The plant was also to be designed for an Operating Basis Earthquake of 0.1g maximum horizontal ground acceleration simultaneously with the other appropriate loads forming the basis of containment design.

COMMENTS ON ADEQUACY OF DESIGN

Dynamic Analyses

(a) Containment Building. The answer to Question 1.9 of the FSAR indicates that only the containment building, the primary auxiliary building, and the electric cable tunnel were designed with the use of semi-formal dynamic analyses. A description of the method of analysis employed is given briefly in Section 5.1.3.8 of the FSAR and in Section 3.1.5 of the containment design report. The procedure employed involved a calculation of the fundamental frequency and mode shape by use of a modified Rayleigh method. The base shear for the structure was computed from the period and the spectral response corresponding to the appropriate degree of damping. The base shear was then applied as a loading to the structure as an inverted triangular loading. The shears at the nodes were used to calculate the moments and displacements at various points in the structure. For the structures involved it is believed that the approach leads to a design which is reasonably adequate.

A similar approach was followed for the primary auxiliary building as described in the answer to Question 1.9. It is noted there that a one-third increase over working stress was allowed in the design of the bracing in the case of the Design Basis Earthquake. This stress is below yield, and it is believed that the design will prove to be satisfactory.

(b) Other Buildings and Equipment. The discussion presented in answer to Question 1.9 of the FSAR for other buildings and equipment such as the control building, fan house, intake structure, etc., indicate that a refined static approach was used, which involves employing the peak value from the appropriate response spectrum curve for a given value of damping and multiplying this by the appropriate mass to obtain the inertial loading. From the description given for the various buildings and items of equipment, and the modeling techniques employed, it is concluded that the inertial loadings used in design are reasonably close to those that might be obtained with a more sophisticated analysis and lead to reasonable design values.

The submission in Question 1.3 of Supplement 13 indicates that the Turbine Building, and Fuel Storage Building Structure above the Fuel Storage Pit were reanalyzed by a multi-degree-of-freedom modal dynamic analysis method to check their adequacy. As a result of this reanalysis, the applicant advises that certain structural modifications will be made to columns and cross bracing in the Turbine Building to insure that it can withstand the DBE. The superstructure of the fuel storage building was ascertained to be adequately designed, without modification to withstand the effects of the DBE. The applicant states that reanalysis of the strengthened turbine building and superheater building for Indian Point No. 1 does not significantly affect the responses calculated for the original structures.

(c) Piping Analysis. The method used by the applicant for analysis of the piping, as described in the answer to Question 1.6 of the FSAR, is the same as was used in Ginna. The peak ground response spectrum value for 0.5 percent damping was used, applied as static accelerations in each direction separately, and the resulting stresses superposed. It was assumed by the applicant that the piping was supported along rigid systems and therefore not subjected to amplified ground motion at points of support. The system was analyzed with the anchors and supports as actually used, according to the discussion presented to us during the time of our visit in May, 1969. It was the view of the applicant that the thermal motions were greater than any differential ground displacements and the latter therefore are not critical items in the design. In answer to Question 1.13 (Suppl. 13) the applicant advises that relative seismic displacement was considered for the main steam lines, where the largest relative displacements are expected; stress differentials of less than 10% resulted. Also, seismic supports installed to date are those specified in the design and employed in the analyses; where deviations in supports must occur, reanalysis will be carried out. These results and approaches appear satisfactory to us.

Since this plant was designed before recent developments and changes in piping design specifications, the 1968 ASME Addenda were not applied. Blow-down and earthquake were considered as separate items and not combined in this design. We are advised that the response to Question 1.9 of Supplement 12 states that a review of the Indian Point 3 reactor coolant system which is identical to Indian Point 2, for combined earthquake and blow-down indicates that the design is adequate.

It is stated in the answer to Question 1.6 of the FSAR that the approach resulted in a seismic design load approximately equal to 0.60W horizontally and 0.40W vertically taken simultaneously. It is further stated that for the Design Basis Earthquake the sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code allowable stresses. In a similar manner the stresses in the pipe supports and hangers were limited to 1.2 times code allowable stresses.

The applicant originally made use of the maximum spectrum value only and no modal analyses were made; in other words only a static analysis with uniform accelerations was made. Consideration was not given to modified distribution of the inertial loading to take account of the combination of modal effects.

The response to Question 1.9 of Supplement 8, describing more detailed analyses of the reactor coolant system, feedwater lines, surge lines and typical steam lines by more formal methods as carried out later lends confirmation to the adequacy of the design. On this basis, there is reason to believe that the design is adequate.

Backfill Surrounding Containment Vessel

Nine feet of crushed rock backfill was placed between the external wall of the reinforced concrete containment vessel and the retaining wall holding back the rock on the uphill side. This crushed rock backfill is drained at the bottom to avoid water pressure against the containment structure. The fill is approximately 60 to 70 feet higher on one side of the structure than on the other because of the slope of the rock surface. The design, as discussed in Section 3.1.5 of the containment design report, considered local inertial forces of loose rock as an added loading against the containment pressure vessel, and also considered passive pressures caused by failure of the rock along the surface behind the retaining wall. The localized loadings from these forces were considered in the design of the containment structure and the discussion presented in the containment design report provides reasonable assurance that the containment vessel is capable of resisting these localized forces.

Class I Equipment in Structures other than Class I

The turbine building is Class III and not designed for earthquake loadings. The answer to Question 1.3 of the FSAR indicates that the only Class I structures and components which are so located that they could be endangered by failure of Class III structures are the control building, main steam piping and feedwater piping, all of which could possibly be endangered by the Class III turbine building. It is further indicated there that no special provisions have been provided for protection except in the case of the main steam and feedwater lines up to the isolation valves, which are protected by the shield wall and the structural frame at the north end of the shield wall. Since these are located near the braced end of the turbine building, it is not anticipated by the applicant that there will be any structural failure in this area. Our judgment as to the adequacy of this aspect of the design is based on the statement given in the application. And, in this respect, the answer to Question 1.3 (Supplement 13) which describes the analysis and strengthening of the Turbine Building and Superheater Building for Indian Point Unit No. 1, and their ability to withstand the DBE, should give additional protection for the control room.

It is further stated that the only Class III crane whose failure could endanger any Class I function is the fuel storage building crane and that the failure of this crane will not impair a safe and orderly shutdown. The answer to Question 1.3 (Suppl. 13) indicates that the only potential for crane lift off will be in the unloaded condition with the trolley parked ...er the support; the applicant advises that the unloaded crane will not be parked over the pool, so no hazard exists. It is also noted in the answer to Question 1.1.3 that the manipulator crane in the containment building, a Class III crane, is restrained from overturning and will not endanger Class I structures.

Deformation Criteria

The general stress criteria applicable to the seismic design are summarized in Appendix A of the FSAR. The statement given on page A3 of Appendix A states that for all components, systems and structures classified as Class I, the primary steady state stresses, when combined with seismic stresses resulting from the response to the Design Basis Earthquake, are limited so that the function of the component system or structure shall not be impaired so as to prevent a safe and orderly shut-down of the plant.

We were advised at the time of our inspection of the plant in May 1969 that, for normal loadings plus the Operating Basis Earthquake, the intention was to use code allowables plus the 20 percent increase for transient conditions on Class I components and systems. For the Design Basis Earthquake and blow-down, basically the same criteria were used, although originally it had been planned to adopt higher allowables going into the plastic range using the code for faulted conditions. In actuality, as described in the answer to Question 1.7 of the FSAR, the allowable stresses in the case of the Design Basis Earthquake were limited to the yield point, or slightly below (see answer to Question 1.3 of Supplement 13).

The only references that we note where there was a calculation of stresses exceeding the yield point were at several places in the containment design report where it was mentioned that the calculations indicate that there could be possible local yielding of the liner under certain loading combinations, but that this would be limited and not be expected to be of a nature as to cause concern with regard to the integrity of the liner.

Reactor Internals

The mechanical design and evaluation of the reactor core and internals is described generally in Section 3.2.3 of the FSAR. From the discussion given it appears that the core support structure and core barrel have been designed with proper attention to support points and limitations of motions. The design criteria for the internals themselves, and specifically with reference to deflections under abnormal operation, are given in Table A.3-2 of the FSAR. These appear reasonable and should provide an adequate margin of safety.

Large Penetrations

A finite element analysis of the large penetrations in the containment vessel was made by the Franklin Institute and a description of the analysis and the results obtained is presented in the containment design report. Several analyses were made for different load combinations, and in addition a number of hand calculations were made to check the order of magnitude of the expected forces and stresses and to verify that the results were reasonable Our review of the material presented, to the extent possible, indicates that the penetration design is adequate.

Splices in Large Reinforcing of Bars

Cadweld splices were used in general in the construction of the containment vessel. We were advised that the early splices, about 10 percent of the total, were made with a bronze base, and the remaining 90 percent were made with ferritic base filler metal. Around the hatch opening, we observed that there was approximately a three foot stagger of adjacent splices, but in questioning we learned that there may not be such a stagger over other areas of the containment vessel. Lack of stagger of adjacent splices could lead to planes of weakness and cause cracking under conditions of over-loading. The pressure tests, however, will reveal any such cracking.

Approximately one in 200 splices was removed for test purposes. This is generally adequate.

Instrumentation and Controls

At the time of the May 1969 visit it was ascertained that the applicant considers the control room as a Class I structure and intends that the housing of it will also be subject to Class I requirements. However, the instrumentation for the control room as well as other instrumentation critical to containment and safe shutdown, has been purchased from the vendors according to applicant's specifications. The answer to Question 1.9 describes the vibration tests employed for selected items of essential equipment; the purpose of these tests is to help demonstrate that little or no difficulty will be expected in the operating characteristics thereof under seismic conditions. Although not absolute proof of acceptability, satisfactory test results certainly help to confirm the adequacy of such instrumentation and control items. Further information on the design and procurement approach for protection system equipment is given in the answer to Question 7.27 (Suppl. 13), and lends confirmation to the approach adopted.

Tornado Loadings

The information contained in Section 3.4 of the containment design report, and the answer to Question 5.7 of the FSAR indicates that the structure is designed for the usual wind loadings. The analyses described in Appendix B of Supplement 6, indicate that the containment building can resist the design tornado. What effect if any that a tornado could have on the control room or other critical facilities is not stated. However, the applicant states that the siding of the control room can resist wind velocities up to 162 mph, and the girts (supporting the panels) will fail at 0.62 psi negative pressure; the building is protected by other buildings on the south and west.

Steel Liner and Containment Vessel

The analyses that have been carried out with regard to the liner are summarized in the FSAR and some additional information is presented in the containment design report. It is our understanding that where bulges of the liners occurred during construction, of less than 2 in., nothing was done to correct the bulges. However, when bulges were 2 in. or greater the liner was pushed back into a position of not more than 2 in. away from its intended position, and additional studs were used to anchor the liner in place. Temporary bracing was employed to hold it in position until the concrete was cast. Because of the foregoing, and since the temperature rise in the lower part of the structure in the liner is reduced by the use of insulating material, it is not expected that the departures from the intended original surface will lead to any difficulties.

Proof Test Procedures and Instrumentation

It is our understanding that a detailed description of the proof test procedures is to be submitted at a later date. At the time of our visit in May 1969 it was proposed by the applicant that strain readings be taken only on the liner around the penetrations. We suggested that additional readings be made which would include diameter changes of the penetrations and other measurements that can be made conveniently and without excessive expense to provide evidence that the design meets the design criteria. Fig. 5.13-4 suggests that such readings will be made. In any event, an interpretative report on the measurements that are taken should be provided and should be correlated with the calculations to provide evidence of validity of the design calculations.

Protection of Pipe Lines for Service Water

We were advised that pipelines for service water are embedded in the ground without any special protection. However, there appear to be alternate lines, although they are generally in the same location and/or trenches. In view of the foundation conditions surrounding the plant, and since there is no indication of previous fault motion or potential faulting, this design approach appears to be adequate. If redundancy in critical water supply is desired, it would be preferable to have separate water lines following independent routes.

Seismograph Installation

The answer to Question 1-1 of Supplement 3 indicates that one seismograph will be installed in the yard area, to provide further evidence of the extent of seismic excitation to which the plant might be subjected if an earthquake occurs. This is acceptable to us.

Containment Design Report

The containment design report, prepared for the applicant by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, has proven to be helpful in arriving at an evaluation of many of the factors inherent in the design. The tables presented are useful in helping to arrive at decisions as to the adequacy of the design; we commend those responsible for the preparation of this summary type material.

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We should like to encourage this type of approach to studies of the containment, structures, piping, equipment and other Class I items. We should like to urge that attention be given also to summaries and tabulation of the most important information, in terms of stresses and deformations, including the sources of the various stress components, how they were combined, and related discussion and explanatory material (including figures) which would lend itself to a much better basis for judgment as to the adequacy of design of nuclear facilities in general.

CONCLUDING REMARKS

On the basis of the information made available to us concerning the Class I structures, piping, reactor internals, and other Class I items, it is our belief that the plant possesses a reasonable margin of safety to meet the original design requirements, including the imposed Design Basis Earthquake loading conditions.

REFERENCES

1. "Final Facility Description and Safety Analysis Report -- Vols. I through V including Supplements 1, 2, 4, 5, 6, 7, 8 and 13," Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., AEC Docket No. 50-247, 1969 and 1970.

2. "Containment Design Report," for Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., prepared by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, March 1969. (Labeled Final Draft)

3. "Adequacy of the Structural Criteria for Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2," by N. M. Newmark and W. J. Hall, August 1966.

W. J. Hall

UNITED STATES
DEPARTMENT OF THE INTERIOR
OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

OCT 1 6 1970

Dear Mr. Chairman:

Pursuant to Section 5 of Public Law 89-605 as amended and other authorizations, we are presenting the views of the Department of the Interior in the matter of the application by the Consolidated Edison Company for an operating license for Indian Point Nuclear Generating Unit No. 2, Buchanan, New York, AEC Docket No. 50-247 (Amendment No. 9). The following comments incorporate those submitted by the Federal Water Quality Administration, the Fish and Wildlife Service and the Bureau of Outdoor Recreation.

The unit under review is the second of three units completed or being constructed at the Indian Point site. We note that applications for construction permits for two more units to be located approximately one mile south of the Indian Point site were made in June 1969.

The Department of the Interior does not object to the issuance of the operating license to the Consolidated Edison Company for Unit No. 2 of the Indian Point Nuclear Power Plant. Our position is based upon the firm commitment by the Company as expressed in its responses to the Atomic Energy Commission that it will meet the water quality standards applicable to the receiving waters and that it will take whatever steps are necessary to mitigate any harmful effects that operation of the plant may have on the fishery resources of the Hudson River and tributary waters.

The Company should be commended for the cooperation it has extended to representatives of this Department during the course of our review. The studies which the Consolidated Edison Company is presently engaged in indicate the Company's concern for the potential damages to the environment that could result from operation of this unit and the others planned at and in the vicinity of Indian Point.

We are pleased to note that the Company has made provisions to open part of its land holdings for compatible public recreation use. We express the hope that the Company's public use plans will be finalized and fully implemented at the earliest possible time.

Consolidated Edison has initiated or participated in a number of studies to determine the effects of both radiological and thermal discharges from the Indian Point reactors upon both the temperature distribution and the aquatic life of the Hudson River through its consultants, Quirk, Lawler and Matusky Engineers, and the Alden Research Laboratories of Worcester Polytechnic Institute. The Company has conducted mathematical studies of the probable temperature in the River and has checked these estimates with hydraulic model studies and actual field studies. In addition, Consolidated Edison has supported several independent but coordinated studies of the micro-organisms and aquatic life in the Hudson River and the probable effects of temperature and salinity changes upon them in the vicinity of the Indian Point Plant.

These studies are continuing and have been and will be helpful in assessing the effects of the Indian Point Unit No. 2 and of the other thermal plants which are proposed for construction on the shores of the Hudson River in the vicinity of Indian Point.

We have been provided information on plans for environmental monitoring of radiological and thermal releases proposed as a part of the operating license application. We understand that the plans for water quality monitoring, including radiological concentrations in the environment in microscopic and macroscopic aquatic life are acceptable to the State of New York. They appear reasonable and are considered generally acceptable to the Department of the Interior.

Through the monitoring programs the Company should have the necessary information to control its activities in a manner that will not violate applicable New York State as well as Federal water quality standards, recommendations of any enforcement conference or hearing board approved by the Secretary or order of any court under Section 10 of the Federal Water Pollution Control Act, and/or other State and Federal water pollution control regulations.

In view of the extensive and valuable fish and wildlife resources in the project area, it is imperative that every possible effort be made to safeguard these resources. Therefore, it is recommended that the Consolidated Edison Company be required to:

1. Continue to work closely with the Department of the Interior, New York State Department of Health, and other interested State and Federal agencies in developing plans for radiological surveys.

2. Conduct pre-operational radiological surveys as planned. These surveys should include but not be limited to the following:

   a. Gamma radioactivity analysis of water and sediment samples collected within 500 feet of the reactor effluent outfall.

   b. Beta and Gamma radioactivity analysis of selected plants and animals (including mollusks and crustaceans) collected as near the reactor effluent outfall as possible.

3. Prepare a report of the pre-operational radiological surveys and provide five copies to the Secretary of the Interior prior to project operation.

4. Conduct post-operational radiological surveys similar to that specified in recommendation (2) above, analyze the data, and prepare and submit reports every six months during reactor operation or until it has been conclusively demonstrated that no significant adverse conditions exist. Submit five copies of these reports to the Secretary of the Interior for distribution to appropriate State and Federal agencies for evaluation.

In addition to the above, the Atomic Energy Commission should urge the Consolidated Edison Company to:

1. Meet with the Department of the Interior, New York State Department of Environmental Conservation, New York State Department of Health, and other interested Federal and State agencies at frequent intervals to discuss new plans and evaluate results of the Company's ecological and engineering studies;

2. Conduct post-operational ecological surveys planned in cooperation with the above named agencies, analyze the data, prepare reports, and provide five copies of these reports to the Secretary of the Interior every six months or until the results indicate that no significant adverse conditions exist;

3. Construct, operate, and maintain fish protection facilities at the cooling water intake structure as needed to prevent significant losses of fish and other aquatic organisms; and

4. Modify project structures and operations including the addition of facilities for cooling discharge waters and reducing concentrations of harmful chemicals and other substances as may be determined necessary.

We appreciate the opportunity to provide these comments.

Sincerely yours,

Walter J. Hickel

Secretary of the Interior

Honorable Glenn T. Seaborg
Chairman, United States
Atomic Energy Commission
Washington, D. C. 20545

APPENDIX H
CONSOLIDATED EDISON COMPANY OF NEW YORK
DOCKET NO. 50-247
FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended Dec. 31		
	1969	1968	1965
Long-term debt	$1,981.6	$1,901.6	$1,711.0
Utility plant (net)	3,793.3	3,583.6	3,169.5
Ratio - debt to fixed plant	.52	.53	.54
Utility plant (net)	3,793.3	3,583.6	3,169.5
Capitalization	3,818.4	3,667.6	3,228.1
Ratio - net plant to capitalization	.99	.98	.98
Stockholders' equity	1,836.7	1,766.0	1,517.1
Total assets	4,069.6	3,845.4	3,387.0
Proprietary ratio	.45	.46	.45
Earnings available to common equity	93.1	95.7	89.9
Common equity	1,210.2	1,139.0	1,072.1
Rate of return on common equity	7.7%	8.4%	8.4%
Net income	127.2	128.5	111.8
Stockholders' equity	1,836.7	1,766.0	1,517.1
Rate of return on stockholders' equity	6.9%	7.3%	7.4%
Net income before interest	198.0	193.9	168.4
Liabilities and capital	4,069.6	3,845.4	3,387.0
Rate of return on total investment	4.9%	5.0%	5.0%
Net income before interest	198.0	193.9	168.4
Interest on long-term debt	84.3	77.0	62.7
No. of times fixed charges earned	2.3	2.5	2.7
Net income	127.2	128.5	111.8
Total revenue	1,028.3	982.3	840.2
Net income ratio	.124	.131	.133
Operating expenses (incl. taxes)	830.5	788.3	668.6
Operating revenues	1,028.3	982.3	840.2
Operating ratio	.81	.80	.80
Retained earnings	426.1	400.9	321.7
Earnings per share of common	$2.47	$2.57	$2.42

	1969		1968	
Capitalization at 12/31	Amount	% of Total	Amount	% of Total
Long-term debt	$1,981.6	51.9%	$1,901.6	51.9%
Preferred stock	626.6	16.4	627.0	17.1
[illegible] stock	1,210.2	31.7	1,139.0	31.0
	$3,818.4	100.0%	$3,667.6	100.0%

Moody's Bond Ratings:	
First Mortgage Bonds	A
Dun and Bradstreet Credit Rating	AaA1

EXHIBIT L

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE, and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	---------- **- NOTES -** 1. Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ---------- Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	---------- **- NOTE -** Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump (level or discharge flow) monitor,

b. One containment atmosphere radioactivity monitor (gaseous or particulate), and

c. One containment fan cooler unit (FCU) condensate flow rate monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required containment sump monitor inoperable.	A.1	- NOTE - Not required until 12 hours after establishment of steady state operation. Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	A.2	Restore required containment sump monitor to OPERABLE status.	30 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump (level or discharge flow) monitor,

b. One containment atmosphere radioactivity monitor (gaseous or particulate), and

c. One containment fan cooler unit (FCU) condensate flow rate monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required containment sump monitor inoperable.	A.1	- NOTE - Not required until 12 hours after establishment of steady state operation. Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	A.2	Restore required containment sump monitor to OPERABLE status.	30 days

EXHIBIT M

EN-28

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE SECRETARY

RELEASED TO THE PDR
9/30/92
date initials

September 18, 1992

SML 10/12-16
#7b

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: SECY-92-223 - RESOLUTION OF DEVIATIONS IDENTIFIED DURING THE SYSTEMATIC EVALUATION PROGRAM

The Commission (with all Commissioners agreeing) has approved the staff proposal in Option 1 of this paper in which the staff will not apply the General Design Criteria (GDC) to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. While compliance with the intent of the GDC is important, each plant licensed before the GDC were formally adopted was evaluated on a plant specific basis, determined to be safe, and licensed by the Commission. Furthermore, current regulatory processes are sufficient to ensure that plants continue to be safe and comply with the intent of the GDC. Backfitting the GDC would provide little or no safety benefit while requiring an extensive commitment of resources. Plants with construction permits issued prior to May 21, 1971 do not need exemptions from the GDC.

The Systematic Evaluation Program (SEP) should be closed. The staff should, however, continue the generic review of the SEP lessons learned and prioritize the issues in the Generic Safety Issue program.

'92 OCT -5 AIO :57

cc: The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
Commissioner de Planque
OGC
OIG

SECY NOTE: THIS SRM, SECY-92-223, AND THE VOTE SHEETS OF ALL COMMISSIONERS WILL BE MADE PUBLICLY AVAILABLE 10 WORKING DAYS FROM THE DATE OF THIS SRM

SUPPLEMENT NO. 1

TO

AEC REGULATORY STAFF SAFETY EVALUATION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY

INDIAN POINT NUCLEAR GENERATING PLANT UNIT 2

DOCKET NO. 50-247

November 20, 1970

Prepared by

Division of Compliance
U. S. Atomic Energy Commission

I. INTRODUCTION

This supplements the Safety Evaluation dated November 16, 1970, prepared by the Division of Reactor Licensing of the Atomic Energy Commission (Commission or AEC) in connection with its review of the application of the Consolidated Edison Company (applicant) for an operating license for Unit 2 of the Indian Point Nuclear Generating Station located in the village of Buchanan, in Westchester County, New York.

The AEC regulatory program is founded in the Atomic Energy Act of 1954, as amended, and on implementing regulations and policies adopted by the Commission. The Congress of the United States has established a system of licensing privately owned and operated nuclear facilities. Inherent in the concept of private activities subject to licensing and regulation by a Government agency is the fact that the licensee is held responsible for meeting the objective of the licensing and regulatory system under the provisions of the Atomic Energy Act of 1954, as amended. The objective of the AEC program is to assure that licensed activities will not be inimical to the health and safety of the public or to the common defense and security.

The Division of Compliance, as an integral part of the Commission's regulatory staff, is responsible for conducting the field inspections of AEC licensees to assure that licensed activities are in compliance with the provisions of AEC licenses; the Atomic Energy Act of 1954, as amended; and the rules and regulations of the Commission. Division of Compliance inspections of nuclear power reactors under construction pursuant to an AEC construction permit provide the principal basis for findings as to the status of completion of facility construction and the conformity of that construction to the requirements noted above.

The Division of Compliance inspection program is conducted from five regional offices with each office having responsibility for the inspection of all AEC licensed activities within an assigned geographical area. The inspection program at Indian Point Unit 2 is the responsibility of the Division of Compliance, Region I office located in Newark, New Jersey. A senior reactor inspector, who reports to the Regional Director, is responsible for supervising the inspection program carried out by the various reactor inspectors. Technical direction of the inspection program is provided by the Division of Compliance Headquarters staff which gives direction to the region with respect to the conduct of inspection activities, gives technical support to the region when required, keeps the region informed concerning inspection experiences in other regions, evaluates adequacy of inspections and inspection results, and maintains liaison with other divisions of the AEC regulatory staff on matters which affect the inspection program.

The program for the inspection of the construction of Indian Point Unit 2 has been carried out primarily by Region I inspectors but, in addition, by Division of Compliance Headquarters staff, by other divisions of the AEC regulatory staff, and by consultants to the AEC. The principal activity of the inspectors has involved periodic inspections at the construction site. These site inspections were conducted at non-regular intervals with the inspection frequency dependent on the activities which were in progress at the site. In addition to site inspections, there were inspections at the shops of major equipment suppliers (vendors). There were also inspections at the offices of the applicant and at its contractors for the purpose of inspecting construction records and procedures and engineering reports related to construction matters.

Division of Compliance inspection personnel are experienced and knowledgeable in the practical aspects of construction and operation of nuclear reactors. In addition to the inspectors, specialists in appropriate fields of engineering and technology, who are assigned to the Division of Compliance Headquarters staff and to other divisions of the regulatory staff, are utilized to assist in special inspections. Further, consultants to the AEC also provide assistance as required. The experience and technical competence of inspection personnel are important factors in the effectiveness of the inspection program.

The Division of Compliance inspection activities were directed toward verifying, on a planned sampling basis, that the licensee carries out his safety responsibilities and that the completed facility would conform to AEC regulatory requirements. Systems and components of the facility were selected for inspection on the basis of the regulatory staff's determination as to their importance to the safe operation of the facility. These inspection activities included the following:

1. Review of the applicant's overall quality assurance and quality control programs and their implementation.
2. Inspection of quality control records such as concrete strength test data, material test reports for plate and piping, supplier certifications for piping, valves and fittings, and nondestructive test records for welding.
3. Observation of construction work in progress; e.g., concrete placement, welding associated with vessel construction or piping installation, equipment alignment and installation, and nondestructive testing.
4. Review of construction procedures; e.g., welding procedures and nondestructive testing procedures.

5. Witnessing the performance of major construction tests such as hydrostatic tests of piping and the pressure test of primary containment.

6. Review of program for functional testing of systems and equipment, including the tests planned, the test procedures, and the test results.

7. Review of preparations for facility operations, including such areas as organization and staffing plans and their implementation, program and procedures for fuel loading and power testing, development of routine operating procedures, maintenance procedures, radiation protection procedures, and emergency procedures.

8. Review of component vendor work in progress, quality control activities and records, and fabrication procedures.

The licensee is required to develop and carry out a comprehensive preoperational testing program. The procedures developed under this program are reviewed by Compliance inspectors and comments are directed to the licensee. The performance of selected preoperational tests are witnessed by Compliance inspectors. The results of the tests and the licensee's evaluations are reviewed by the inspectors. This testing of the plant, to the extent possible prior to the loading of fuel, demonstrates whether plant systems and components are capable of performing their intended functions under both normal and abnormal conditions. These tests also serve to demonstrate the adequacy of plant design and operating procedures. Satisfactory completion of the pre-operational testing program is an important part of the basis for our findings of plant completion.

II. RESULTS OF CONSTRUCTION INSPECTIONS

Since the issuance of Provisional Construction Permit No. CPPR-21 to the applicant authorizing construction of Indian Point Unit 2, inspections by the Division of Compliance have been conducted at the construction site, at vendor shops, and at the applicant's offices. A chronology of these inspections is attached as Appendix A. The results of the inspection of Unit 2, conducted through October 14, 1970, are discussed by systems in the same order as presented in the Safety Evaluation dated November 16, 1970, prepared by the Division of Reactor Licensing.

A. Reactor Coolant System

1. Reactor Coolant Pressure Piping

The reactor coolant pressure piping includes the four primary recirculation loops, the pressurizer lines and portions of the following systems: Chemical and Volume Control, Emergency Core Cooling (ECCS), Shutdown Cooling, Safety and Relief Valves, and Reactor Coolant Vent and Drain.

Our inspection program was directed primarily toward auditing fabrication, erection, and nondestructive testing of the reactor coolant pressure boundary components and piping. The effort included site and vendor inspections utilizing our staff specialists. The hydrostatic test of the reactor coolant boundary at 125% of design pressure, which is required by the American Society of Mechanical Engineers (ASME) Code, has been conducted. Portions of this test were reviewed and witnessed by Division of Compliance inspectors and records of test results were examined to assure compliance with the code. In addition to the normal quality control inspections, a special quality control inspection was performed, under the direction of the assigned inspector, by a team of staff specialists, a specialist from the Division of Reactor Licensing, and a consultant. Segments of the reactor coolant system and emergency core cooling system (ECCS) were selected for inspection and review. Material certifications for selected portions of the reactor coolant system components were examined.

Onsite quality control records for the reactor coolant and ECCS systems were examined and visual inspections of these systems were performed. Followup inspections have been made to the site to complete the record review, and at the vendor shop which fabricated the ECCS piping.

The applicant and his contractor performed a review of quality control records for all pipe, valves, and fittings within the reactor coolant pressure boundary. This review confirmed the Division of Compliance findings that the reactor coolant system piping had not received the full hydrostatic test required by the applicable American Society for Testing Materials (ASTM) Code prior to leaving the manufacturer's shop and that certain cast valve discs (7) had not been radiographed. The subsequent performance of a field hydrostatic test of the system is considered to fulfill the code requirements. The necessity for radiographing the discs of the seven valves which do not perform a primary isolation function is being evaluated by the Division of Compliance and the Division of Reactor Licensing.

Completion Status: Construction of the primary coolant piping is essentially complete. Some installation of insulation and pipe hangers remains.

2. Reactor Vessel

The reactor pressure vessel was fabricated at the shops of Combustion Engineering, Inc., in Chattanooga, Tennessee.

The Division of Compliance performed inspections at the shops during which fabrication practices were observed, material quality records were examined, and nondestructive testing methods were reviewed. We have followed the placement of the vessel and fitup of the reactor core internals and installation of the internals vibration detection instrumentation. No deficiencies were identified.

Completion Status: Construction of the reactor pressure vessel and core internals has been satisfactorily completed.

3. Steam Generators

Compliance performed a vendor inspection at the steam generator manufacturer's plant.

This inspection included a review of quality control programs and related essential documentation. The inspection disclosed records which indicated that insulation nut plate welds on the channel heads of the steam generators had not been magnetic particle tested. Subsequent magnetic particle testing of the welds was performed in the field. The Division of Compliance reviewed fitup and girth welding of the steam generators in the field. This activity included a review of welding procedures, welder qualifications, and weld material certification.

Completion Status: Construction of the four steam generators has been satisfactorily completed.

4. Reactor Coolant Pumps

The reactor coolant pumps have been installed and have received an initial operation checkout. We verified the pump materials and nondestructive testing performance for the reactor coolant pumps during the special quality control inspection referred to in paragraph II. A. 1. of this report.

Completion Status: Construction of the reactor coolant pumps has been satisfactorily completed.

5. Pressurizer

The pressurizer has been installed. We reviewed installation of the vessel and verified that the code stamp indicated construction to applicable codes and regulatory requirements. During pre-service ultrasonic testing of the pressurizer welds, nonmetallic inclusions in the base plate material were detected. The applicant conducted additional nondestructive testing and technical reviews pertaining to the existing condition and concluded that a series of nonmetallic inclusions exist within the base plate material and that laminar defects beyond that allowed by the ASME Section III code do not exist. The applicant has submitted a report on this subject to the Division of Reactor Licensing. The acceptability of these nonmetallic inclusions is under evaluation by the Division of Compliance and the Division of Reactor Licensing. This issue will be resolved prior to licensing.

Completion Status: Construction of the pressurizer has been completed; however, satisfactory resolution of the above base plate material question will be required prior to licensing.

6. Pressure Relief and Safety Valves

We have verified that the pressure relief and safety valves were installed and were set at the vendor shop to relieve at the designated pressure.

Completion Status: Installation of these valves has been satisfactorily completed.

Conclusions: Based on the results of previous inspections and corrective actions taken by the applicant and contractor to date, we conclude that there is reasonable assurance that the reactor coolant system will be completed in accordance with AEC regulatory requirements.

B. Containment and Class I Structures

1. Primary Containment

The primary containment is a steel-lined reinforced concrete structure which houses the reactor coolant system. Our inspection program included selective examination of field fabrication procedures, observation of field fabrication activities, observation of non-destructive testing, and selective examination of onsite quality control records.

Problems identified by the applicant during construction of the primary containment included:

a. A marked reduction in cadweld yield strengths was encountered.

b. The nominal diameter of the liner exceeded tolerance limits in some instances.

c. Documentation on pipe penetration bellows materials and weldment quality is only partially traceable.

The applicant and his contractors investigated and resolved to our satisfaction problem a. and b. described above, and have initiated programs for correcting item c. Division of Compliance inspectors followed the progress of the completed investigations during inspections by the applicant at the site, and will follow those that are continuing for item c.

<u>Completion Status</u>: The system will be considered complete following concrete closure of one construction access opening, resolution of

the penetration bellows question, completion of the integrated leak rate test, and installation of the reactor coolant system leak detection equipment.

2. Other Class I Structures

Other Class I (seismic) structures at Unit 2 include the primary auxiliary building, the control room, the fuel storage pool, diesel generator building, and the service water intake structure. Vacuum testing revealed leakage at the welds of the fuel storage pool liner. The applicant and contractors have taken appropriate corrective actions. We have inspected the construction of the other Class I structures from the standpoint of construction practices and concrete quality. No problems were identified.

Completion Status: Construction of the other Class I structures is nearing completion. Items to be completed prior to licensing are:

a. Additional reinforcement of the Unit 1 superheater building (required because of Unit 2 considerations) and the Unit 2 turbine building.

b. Installation of a second completely independent turbine overspeed control.

c. Provisions for alternate charging pump cooling water.

d. Added missile protection for the auxiliary feedwater lines.

<u>Conclusions</u>: Based on our inspections to date, we conclude that there is reasonable assurance that the containment and other Class I structures will be completed in accordance with AEC regulatory requirements.

C. <u>Engineered Safety Features</u>

1. <u>Emergency Core Cooling System (ECCS)</u>

The emergency core cooling system is comprised of a high pressure system, a residual heat removal system, a recirculation system, boron injection tanks, and pressurized safety injection accumulators. We have inspected the construction and examined quality control records for the ECCS during our normal inspections and the special quality control inspection. Results of our inspection included the following:

a. Welding quality control records incomplete.

b. Visual inspection indicated a weakness in first line quality control; i.e., weld splatter, arc strikes, and excessive grinding.

c. Accumulator check valves which were not manufactured to Westinghouse specifications.

The applicant and contractor initiated corrective actions for these items and resolution of each is nearing completion. These items will be reviewed by the Division of Compliance to assure satisfactory resolution prior to licensing.

The applicant and his contractor performed a review of quality records for all pipe, valves, and fittings included in the reactor coolant pressure boundary, as described in paragraph II. A. 1. above. In addition, the applicant has reviewed quality control records for the remainder of the piping included in the ECCS system. The Division of Compliance has audited the results of this review and considers the findings to be acceptable.

<u>Completion Status</u>: Construction of the ECCS system is essentially complete. Remaining work

to be accomplished includes: (1) finish surface cleanup, (2) completion of hanger and support installation, and (3) resolution of items listed above.

2. Containment Spray and Fan Cooling Systems

The containment spray system is comprised of two spray pumps and chemical additive devices. We have inspected the construction and examined quality control records for this system in conjunction with the ECCS.

The containment fan cooling system is located within the containment. The Division of Compliance plans to complete inspection of this system during functional testing and filter testing prior to licensing.

Completion Status: Construction of the containment spray and fan coolers is nearing completion. Work remaining includes filter testing and functional testing.

3. Post Accident Hydrogen Control System

The post accident hydrogen control system has not been installed. Installation of this system will be verified when completed.

Completion Status: Installation of the hydrogen control system will be completed prior to licensing of Unit 2.

Conclusions: Based on the results of our inspections to date, we conclude that there is reasonable assurance that the construction of the Engineered Safety Features will be completed in accordance with AEC regulatory requirements.

D. Instrumentation, Control, and Power Systems

These systems include the reactor protective, control, safety, and nuclear instrumentation and normal and emergency power. We have inspected the quality of the electrical and instrumentation installation, the separation and protection of key safety related circuits, and the loading of cable trays and wireways during the course of our normal inspection and, also, during the special quality control inspection. Our inspection observations included the following:

1. Independent cable design review had not been performed.
2. Independent quality control of cable installation was lacking.

3. Some redundant cables were not properly separated.

4. Items which required additional design analyses.

The applicant and contractor initiated responsive actions to correct the conditions noted above. Compliance has verified that their actions included a 100% design audit relative to the separation of power and control electrical cabling for redundant engineered safety feature and a design review on associated instrument cabling in excess of 95%. We have verified that work on the remaining items listed above is nearing completion. These areas will require additional Compliance inspection effort to assure satisfactory completion prior to licensing.

Completion Status: Construction of the electrical and instrumentation systems is 95% complete. Items remaining to be completed include:

1. Installation of remainder of separation barriers and fire stops.

2. Completion of cable installation surveillance program.

3. Installation of transite barriers at the single penetration area.

4. Installation of redundant power cables for the tunnel fans.

Conclusions: Based on the results of previous inspections and corrective actions taken by the applicant and contractor to date, we conclude that there is reasonable assurance that the instrumentation, control, and power systems will be completed in accordance with AEC regulatory requirements.

E. Radioactive Waste Control

The radioactive waste control system includes facilities for processing and minimizing releases of liquid and gaseous effluents to the environment. We have inspected the installation of the major components of these systems. The radiation monitoring instrumentation has not been installed and will be inspected for acceptable installation prior to licensing.

Completion Status: The radioactive waste control systems are essentially complete with the exception of the radiation monitoring instrumentation and controls.

Conclusions: Based on inspections to date and the applicant's planned actions, we conclude that there is reasonable assurance that the radioactive waste disposal system will be completed in accordance with AEC regulatory requirements.

F. Auxiliary Systems

Auxiliary systems include chemical and volume control, residual heat removal, component cooling service water, and spent fuel storage.

Completion Status: Construction is essentially complete. Work to be accomplished includes installation of some insulation, hangers and supports.

Conclusions: Based on the results of inspections to date, we conclude that there is reasonable assurance that the auxiliary systems will be completed in accordance with AEC regulatory requirements.

G. Conduct of Operation

Conduct of operation as used here includes organization and staffing, preparation and review of procedures, and the administrative directives which the applicant has developed to conduct the functional testing program and subsequent operation of the Unit 2 facility. We have verified that the applicant has established operational review and audit committees which are actively engaged in activities relating to plant startup. We have verified that the applicant has developed a program for functional testing of equipment and systems and we have examined the available test procedures on a selective basis. We have also selectively examined the results of tests which have been completed. We have initiated our review of the program and procedures for fuel loading, power ascension testing, and plant operation. We plan to examine these procedures on a selective basis when their preparation has been completed.

<u>Completion Status</u>: Sixty percent of the preoperational test procedures have been approved for use by the applicant. System functional testing is in the initial stages. Preoperational testing, including hot functional testing is scheduled to be completed prior to licensing.

<u>Conclusions</u>: Based on the results of our inspection to date and responsive action taken by the applicant previously, we conclude that the administrative organization is in conformance with the application and that testing will be completed in accordance with AEC regulatory requirements.

III. CONCLUSIONS

Based on the results of inspections of the Indian Point Unit 2 facility, we conclude that construction of the facility has been substantially completed in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

APPENDIX A

CHRONOLOGY OF COMPLIANCE DIVISION INSPECTIONS
CONSOLIDATED EDISON COMPANY
INDIAN POINT NUCLEAR GENERATING STATION UNIT 2

Date	Type Inspection	Scope of Inspection
5/10-12/66	Shop Inspection - Combustion Engineering, Chattanooga, Tennessee	Inspected shop facilities and discussed procedures for fabricating the reactor vessel.
11/2/66	"	Reviewed fabrication progress of reactor vessel. Observed work in progress and discussed fabrication techniques.
5/2/67	Site Inspection Management Meeting	Initial meeting with Con Ed management to discuss Division of Compliance inspection program during reactor construction.
5/24-26/67	Shop Inspection - Combustion Engineering, Chattanooga, Tennessee	Reviewed fabrication progress, observed work in progress, and inspected records of welding, plate material properties and radiography.
8/1, 16, 22/67	Site Inspection	Reviewed construction organization responsibilities. Inspected containment liner installation. Reviewed quality control program for concrete, reinforcement bar and containment liner activities. The program relating to blasting control was discussed.
11/29-30/67	Site Inspection	Reviewed corrective actions on containment liner bulge. Inspected records on containment liner plate and reinforcement bar materials. Reviewed cadweld splice quality control program and information relating to decrease in cadweld strengths. Inspected concrete compressive strength results. Reviewed blasting control program.

Date	Type Inspection	Scope of Inspection
2/27-28/68	Site Inspection	Reviewed quality control records on cadweld splicing, concrete, containment liner and blasting. Reviewed quality assurance program relative to procurement of off-site components.
4/22-24/68	Vendor Inspection - Combustion Engineering, Chattanooga, Tennessee	Reviewed records of reactor vessel fabrication. Witnessed initial closure of reactor vessel head and hydrostatic testing of the vessel.
3/14/68	Site Inspection	Reviewed quality assurance programs and availability of records for procured components.
6/17-18/68	Site Inspection	Inspected containment liner, cadweld splice, concrete, and blasting records. Reviewed the spent fuel storage liner installation. Inspected steam generator components and reviewed photographs of the steam generator movement from the barge to the site.
6/19/68	Site Inspection	Reviewed vendor inspection reports for procured components. Reviewed purchase specification for the steam generators and the safety injection accumulators.
7/8-9/68	Vendor Inspection Chicago Bridge & Iron, Greenville, Pennsylvania	Reviewed purchasing, quality control, production, and records control for fabrication of the containment liner.
9/27 and 30/68	Site Inspection	Reviewed records pertaining to the containment liner, cadweld splicing and concrete. Reviewed the material receipt inspection program and welding procedures for the safety injection system. Inspected component storage areas. Visually observed the conditions relating to the steam generators and reactor vessel. An initial review of training and preoperational testing was made.

Date	Type Inspection	Scope of Inspection
10/8/68	Site Inspection	Reviewed electrical design criteria relating to cable sizing and tray loading.
11/20-21/68	Site Inspection	Reviewed testing records for cadweld splicing and concrete activities. Reviewed actions taken to resolve quality deficiencies in the conventional and safety injection system pipe. Inspected the reactor vessel, steam generators, and reactor coolant pumps for visible deficiencies.
1/7-9/69	Vendor Inspection - Dravo Corporation, Marietta, Ohio	Inspected fabrication and quality control records pertaining to pipe procured.
1/20 and 24/69	Site Inspection	Reviewed cadweld splicing and concrete test records. Inspected records and procedures pertaining to field fabrication of the reactor coolant system and the steam generator girth welding. Reviewed resolution status of identified conventional pipe deficiencies. Observed machining of the reactor vessel lower internal supports and electrical installation.
3/4-5/69	"	Reviewed records pertaining to cadweld splicing and reactor coolant system welding. Inspected safety injection system weld records and field conditions. Observed steam generator fitup and girth welding and reviewed associated records. Inspected external storage of components.
3/18-21/69	Vendor Inspection - Westinghouse Electric Corporation, Lester, Pennsylvania	Reviewed quality control programs and essential documentation for the steam generators.

Date	Type Inspection	Scope of Inspection
4/22-23/69	Site Inspection	Reviewed pipe specifications, vendor assembly records, storage and installation as related to investigations of piping fabricators' practices.
4/17 and 5/15, 22, 23/69	"	Reviewed quality control records for cadweld splicing, reactor coolant system welding, safety injection system site erection, and the spent fuel pit liner. Reviewed actions taken relative to safety injection and conventional system pipe component deficiencies. Inspected revised steam generator girth weld procedures and records relating to this activity. Reviewed activities associated with pipe fabrication investigations.
6/17, 7/1-2/69	"	Inspected quality control records for cadweld splicing, concrete placement, and welding for the reactor coolant and safety injection systems. Reviewed electrical cable placement control programs and status of investigation relating to pipe procurement. Inspected pipe supports, component outside storage and code stamping of components.
7/23-24/69	Site Inspection	Reviewed progress relating to resolutions pertaining to pipe investigation. Inspected portions of the safety injection system mechanical components to determine proper physical arrangements. Reviewed welder and weld procedure qualification and welding performance for the control rod vessel head seal welds.

Date	Type Inspection	Scope of Inspection
8/26, 27, 29/69 and 9/10/69	Site Inspection	Reviewed the status of the pipe investigation and the proposed organizational changes relating to the establishment of the Wedco, Inc. subsidiary of Westinghouse. Observed reactor coolant system welding. Inspected the electrical cable placement and separations programs. Reviewed the physical layout and preoperational checkout of the fuel storage building. Reviewed procedures for fuel element receipt and storage.
9/30/69 and 10/1-2/69	"	Continued the review of the pipe investigation. Reviewed welding records for the reactor coolant and safety injection systems. Inspected electrical cable placement progress and conformance to separation criteria. Observed the initial receipt and handling of fuel assemblies. Reviewed records relating to containment liner installation at the construction access openings. Reviewed reactor vessel nozzle weld overlay procedures. Observed attachment of reactor vessel internals vibration detectors and control programs for the vessel internals.
12/9-19/69	Quality Control Audit at the site, Con Ed Engineering offices, and Westinghouse Electric Company at Monroeville and Cheswick, Pennsylvania.	Team inspection to evaluate quality control of preselected portions of the reactor coolant, safety injection, main steam, and electrical systems.
2/10/70	Management Meeting	Discussed results of quality control audit performed in December 1969.

Date	Type Inspection	Scope of Inspection
1/22/70 and 2/6 and 11/70	Site Inspection	Reviewed final status of pipe investigation. Inspected the general preoperational test program and initial portions of system flushing and hydrostatic testing procedures. Reviewed conformance to reactor pressure boundary criteria for installed components.
3/26-27/70	"	Continued inspection of preoperational testing program. Reviewed placement and surveillance activities for electrical cables, placement of cadwelds at the containment construction access openings, and status of resolution of items identified during the Quality Control Audit.
4/10, 21, 22/70	"	Continued inspection of the preoperational test program, electrical cable placement, and containment closure. Reviewed the proposed operating organization and status of operator training. Reviewed installation of vibrational detection instrumentation for the core internals.
5/6-8/70	"	Continued inspection of preoperation test program, electrical installation and containment closure. Reviewed status of mechanical surface cleanup.
5/22, 25, 26/70, 6/3, 11, 12, 15, 16/70	"	Continued inspection of the preoperational testing program, electrical installation control programs, mechanical systems cleanup review, and evaluation of reactor pressure boundary components. Made initial inspection of radiation monitoring and waste handling systems.

Date	Type Inspection	Scope of Inspection
6/26 and 29/70, 7/8-9/70	Site Inspection	Witnessed the reactor coolant system hydrostatic test. Continued inspection of preoperational test programs, electrical installation reviews, and previously identified and unresolved items. Made initial inspection of the operating procedure program and nuclear facility safety committee structure and involvement. Reviewed status of previously identified items requiring resolution.
7/30/70 8/4, 5, 19, 24, 25/70	"	Continued inspections of preoperational test programs. Reviewed status of electrical installation, mechanical systems cleanup, reactor pressure boundary, and containment closure activities. Reviewed conditions noted during preservice UT inspection of the pressurizer.
9/8, 23, 25/70	"	Continued inspection of preoperational test program, mechanical system cleanup & containment closure activities. Reviewed installation control programs for pipe supports. Examined ultrasonic test data for the pressurizer base plate material.
10/7, 8, 13, 14/70	"	Continued inspection of the preoperational testing program, mechanical system cleanup, containment closure, and pipe support installation. Reviewed pipe penetration bellows welding and materials documentation. Continued inspection relating to reactor pressure boundary components, electrical design reviews, and electrical cable placement surveillance. Reviewed organization and involvement of the Nuclear Safety Committee. Continued evaluation of the pressurizer base plate material. Reviewed status of previously identified items requiring resolution.

EXHIBIT N

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Protection against any action of the engineered safety features which would accentuate significantly the adverse after-effects of a loss of normal cooling shall be provided.

> The intent here was simply to state the criterion in a more positive way.

CRITERION 44 - EMERGENCY CORE COOLING SYSTEM CAPABILITY (Category A)

An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

> In our opinion, one emergency core cooling system which incorporates a sufficient redundancy of active components and covers the full range of postulated breaks should be adequate. Our modification of this criterion reflects this consensus. For this reason, we have omitted the last sentence of the original criterion.

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM (Category A)

Design provisions shall where practical be made to facilitate physical inspection of all critical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles.

> Since inspection of water injection nozzles is not always possible on a reasonably complete and non-destructive basis and since the failure of a safety injection nozzle is assumed in most accident analyses, we have inserted the phrase, "where practical".

CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEM COMPONENTS (Category A)

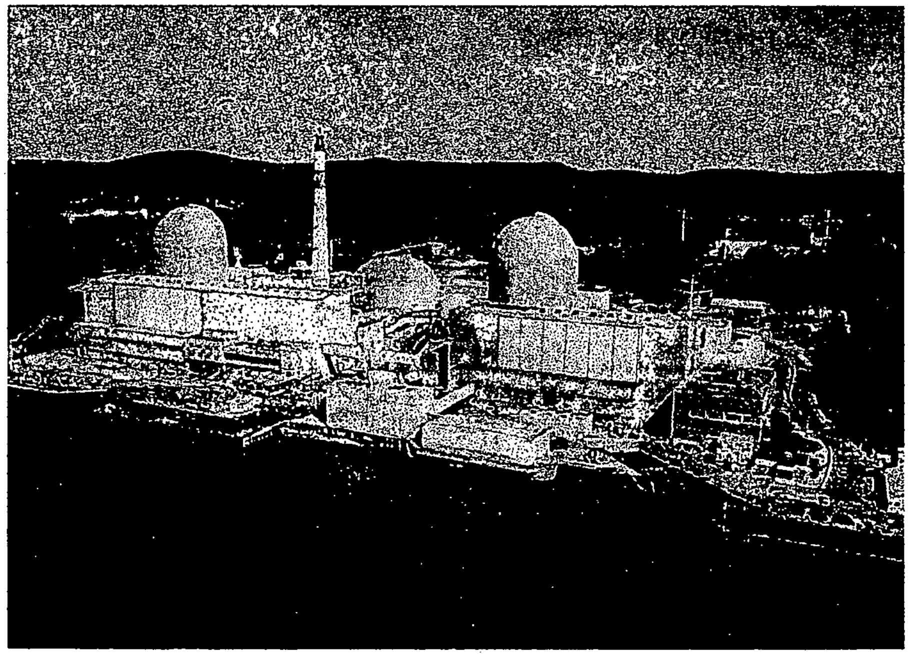
> No comment other than the criterion should be presented in the context of a single emergency core cooling system, consistent with the comments offered on Criterion 44.

EXHIBIT O

EXHIBIT P

Indian Point Units License Renewal

Indian Point Unit 1, 2 and 3



Presentation by Karl Jacobs

Background

- Local Resident of Cortlandt Manor for 18 years
- Have never been in the employ of Entergy
- 20 years of experience with nuclear operations, maintenance, project management, installation of major multi million dollar safety related nuclear and non- nuclear equipment at IP3 that meets the required federal, state and industry accepted codes.
- 20 years of experience primarily on Indian Point Unit 3 in developing and implementing aging management programs for the Reactor Vessel, Reactor Internals, Pressurizer, Rector Coolant Piping and Steam Generators etc.
- Participated in the License Renewal rulemaking (10CFR50.54a) as IP3 Utility representative and as a Westinghouse Owners Group (PWR NSSS) Subcommittee Chairman, Nuclear Energy Institute (NEI), Electric Power Research Institute and the Nuclear Regulatory Commission
- Lead Technical Engineer for the technical and economical studies for Indian Point Unit 3 and James A. Fitzpatrick Nuclear Plant License Renewal evaluations. The IP3 studies were performed for the previous owner are identified.
  - License Renewal Comparison of IP3 design, operation and performance characteristics to the Industry Pilot Plant (Surry 1) .
  - Life Extension/ License Renewal Program Technical Summary Report
  - Cost/Benefit Analysis

Highlights of the 10CFR 50.54 and revised 10CFR51 Rule

Identification of the License Renewal Components for scoping and screening evaluations and if determined technically that a component does not meet the additional life extension requirements (an aging management programs would be identified for implementation (on –going current licensing basis programs, newly developed and required to be implemented during their license renewal period)

- **This scoping is also to include the identification and evaluation of time limited aging analysis (TLAA)**

Environmental Impact Studies – Opens the door for Cooling Towers to be evaluated and possibly installed in lieu of present Water Cooled Condenser System – The Cooling Towers would help address the zebra mussel issues which are an environmental issue that in the past has plagued the safety related service component and service water cooling systems for IP3 and IP2. (Reduction and possible removal of their chlorination injection program, will also benefit the Hudson River.)

Identify and /or develop aging management programs of the components that are identified through the screening process for managing aging effects and address TLAAs

Emergency Planning and Security is not part of the 10CFR50.54 and revised 10CFR50.51 rule and needs not to be addressed under License Renewal Application

Indian Point Unit 1 License Renewal Scoping Issues

The license renewal application (LRA) is for IP2, IP3 and shared systems with IP1

A review of the scoping of components in the LRA the does not identify Indian Point Unit 1 Containment structure and spent fuel systems and their support systems as being part of the License Renewal Application. See LRA Section 2.4.1 Describes only Unit 2 and Unit 3 Vapor Containment Structure. Unit 1containment structure is omitted.

Per the License Renewal Application for IP2 and IP3 under containment scoping and screening review in section 2.4.1 page 2.4.-2 state "the containment buildings have the following intended functions for 10CFR54.4(a)(1), (a)(2) and (a)(3)."

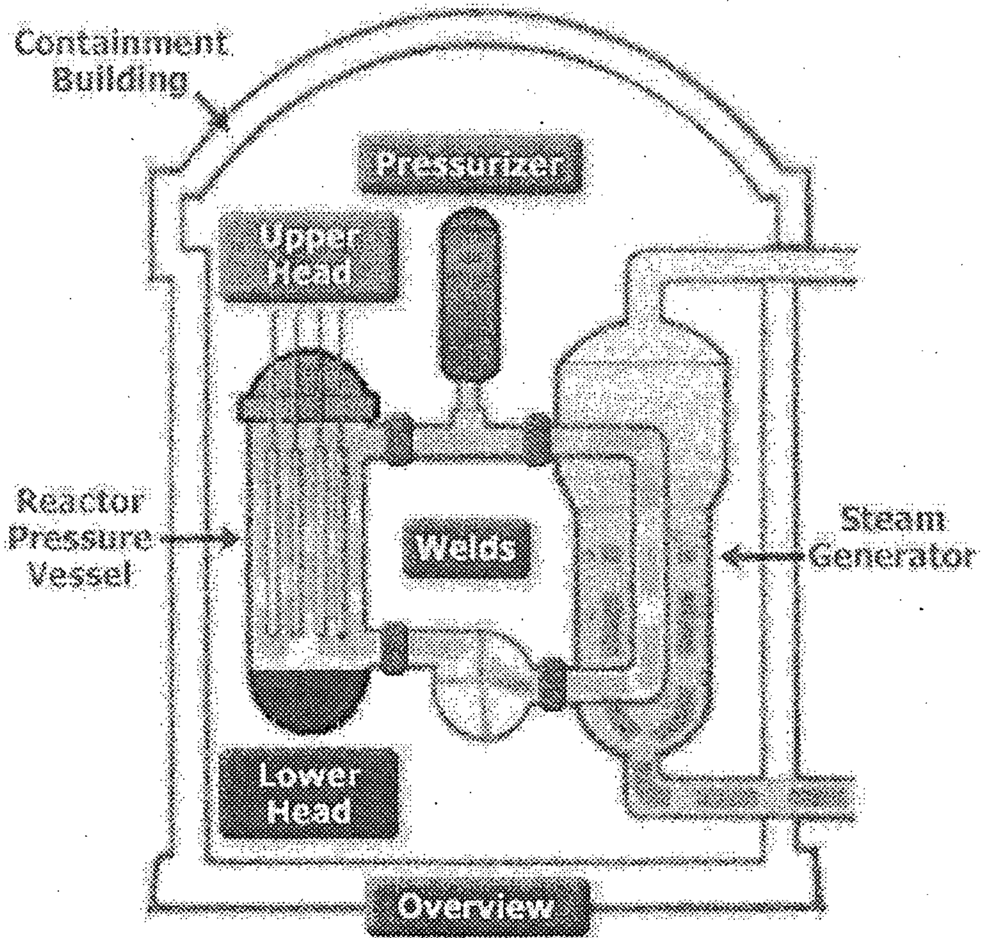
- **Provide support, shelter and protection for safety- related equipment**
- **Maintain essential leak tight barrier**
- **Maintain integrity such that safety –related equipment is not affected.**

Indian Point Unit 1 License Renewal Scoping Issues

Indian Point Unit 1 supports the spent fuel cooling system is located in the containment structure. IP Unit 1's containment performs intended functions as defined by the License Renewal rule function above. In addition other scoping of license renewal scoping and screening systems are inside the containment structure that have been excluded are spent fuel pools structures; HVAC filtration for radioactive airborne particulates, containment penetrations, spent fuel pool system cooling piping and their supports, spent fuel cooling pumps, instrumentation for monitoring the operations of the spent fuel system, electrical wiring, spent fuel bridge,cranes and radiation monitors etc.

With the Entergy IPEC LRA allowing for IP Unit 1 shared components to be included in their application has opened a doorway to allow for a full scoping and screening of IP Unit 1 systems and components to protect the health and safety of the public

IP2 and IP3 Typical RCS Integrity Boundary



Reactor Vessel and Reactor Internals Typical to IP2 and IP3

Westinghouse Nuclear Steam Supply System Designer and Fabricator of Reactor Internals

Combustion Engineering is the Reactor Vessel Fabricator

IP2 RPV Construction Code – ASME Section III 1965 Edition

IP3 RPV Construction Code – ASME Section III Edition Winter 1965 Addenda

Reactor Vessel (RPV)

- **Reactor Vessel Major Intended Functions**
  - Maintain the reactor pressure boundary
  - Support and contain the reactor core and core support structures
  - Support and guide reactor controls and instrumentation
  - Contain the reactor coolant around the reactor core and direct the coolant flow into the core and out into the reactor coolant piping and upper head
  - Interface with the RPV supports to provide a load path to the structural concrete
- **Subcomponents subject to an aging management review**
  - All of its subcomponents are passive, and only two of the subcomponents do not require an aging management.
  - There are only two subcomponents that do not require an aging management review. The RPV O-Rings, O-ring leak monitoring tubes and the refueling seal ledge do not support any RPV intended function

Reactor Vessel (RPV)

- For RPV neutron embrittlement is a critical aging management failure mechanism issue that must be accurately evaluated for License Renewal for both IP2 and IP3 reactor vessels.
- This IP3 reactor vessel has a projected RTndt value that would have exceeded the 10CFR50 Appendix G criteria during life extension if the criteria was not revised by the NRC
- For IP3 the lower shell plate (B2803-3) is the limiting RPV plate material.
- The projected RTpts for this same lower shell plate is very close to the 10CFR50 Appendix G criteria for the end of license renewal. With augmented aging management programs being implemented which are low leakage fuel management for neutron flux reduction,significant expansion of the reactor vessel surveillance capsule monitoring program, implement research and development programs on material crack initiation and crack growth with similar low fracture toughness' properties, along with a higher frequency of volumetric examinations of the RPV beltline than the present frequency requirements of ASME Section XI and Regulatory Guide 1.150 the RTpts may be successfully managed to meet life extension.
- For the same plate, the projected upper shelf fracture toughness energy for 60 calendar years is less than 10CFR50, Appendix G minimum criteria of 50- ft-lbs. This is a critical issue, that Entergy will need the NRC's assistance in a 10CFR50 Appendix G rule change to revise the criteria to a lower threshold value. This plate was originally installed with an initial +74 RTndt value. This was a fabricator miscue to allow the original installation of a shell plate in the Reactor Vessel Beltline with a +74 RTndt material property value to be installed. The plates that are installed in reactor vessels should have minimum initial Rtndt value of zero or a minus value to support Reactor Vessel longevity.

RPV

- The IP3 Reactor Vessel's lower upper shelf energy (a physical/mechanical properties of the RPV vessel wall) is a major concern for its lower shell plate B-2803-3. This plate material will not meet 10CFR50 Appendix G "Fracture Toughness Requirements" for license renewal. This plate is predicted to fall well below the 50 ft-lbs as measured by Reactor Vessel Surveillance Capsules charpy v –notch specimen testing.
- IP3 has two alternative approaches which are not even mentioned.
    1. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation. The margins against fracture must be equivalent to those required by the ASME Code, Section III, Appendix G
    2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests. The problem with this approach is the IP3 Reactor Surveillance Program remaining capsule specimens do not have the limiting plate material B2803-3 in any of this capsules. The statement made by Entergy in the license renewal application Section B.1.32,titled (Reactor Vessel Surveillance) page B-112 under the described enhancements that "The specimen capsule withdrawal schedules will be revised to draw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation."

RPV

- **The IP Unit 3 RPV has an on going aging management program called reactor vessel surveillance capsule monitoring program that is in effect for its current licensing basis and does an extra capsules installed in the reactor vessel for life extension. The limiting plate material (B2803-3) as does not have any material in the remaining capsules to monitor the lower shell plate. This also will have a significant effect on their heat and cool down curves which are developed from the most limiting vessel plate material.**
- **The IP3 RPV materials has been volumetric examined (ultrasonic techniques) thoroughly every ten years from initial operation and no reportable indications were found. The volumetric equipment used for these inspections are very good, use a array of ultrasonic transducer probes, with high detection capabilities, sizing and locating any flaws are also very good. Please note that the RPV beltline is 100% inspected but access to allow for 100% of all RPV welds volume is not achievable do to interferences.**
- **Bottom line IP3 reactor vessel beltline plate material absolutely does not support license renewal unless the NRC revises 10CFR50 Appendix G requirement of maintaining a higher USE value of 50 ft –lbs for its belt line material**

RPV

Present Industry Events and experiences has identified that the IP2 and IP3 Reactor Vessels' Heads must be replaced prior to life extension. This is a generic industry concern for the Westinghouse Reactor Vessel Heads' penetration tube welds that started in September 1991 @ the Bugey Unit 3 PWR nuclear plant in France. Then in May 1992 Ringhals Unit 2 , a Westinghouse- designed PWR –in Sweden found a 25 % around through wall crack in the CRDM penetration. Then it came to America. 1995 DC Cook Unit 2 (Westinghouse design) a crack measured as the deepest point of 6.88mm, 25% around the CRDM tube wall. VC Summer Plant was next, then Ringhals 3 and 4in June 2001, then Oconee and an Entergy Plant ANO-1. NRC Bulletins have been issued.

- NRC Bulletins 2001 –01 Circumferential Cracking of Reactor Pressure Head Penetration Nozzles
- NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity
- NRC Bulletin 2002-02 Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs

– Entergy LRA response – Intend to use ASME Section XI, Sub Section IWB Inservice Inspection and Water Chemistry Control Programs. Detection of Cracking is accomplished through implementation of a combination of bare metal visual examination (external surface of head) and non-visual examination (underside of the head) techniques.

– Entergy has not realized as a company that safety and lowering the risk to public health comes first not economics This is real cracking issue that many same design plants are experiencing now! This cracking can lead to a control rod missile ejection followed with a small break loca. This failure would permanently shut IPEC down!

– Reactor Coolant Supports are located in a difficult to access area and limits inspection capabilities. Reactor Coolant Supports can corrode since the are serviced with cooling water. A inspection program to fully assess these reactor supports and cooling system requires a definitive aging management program.

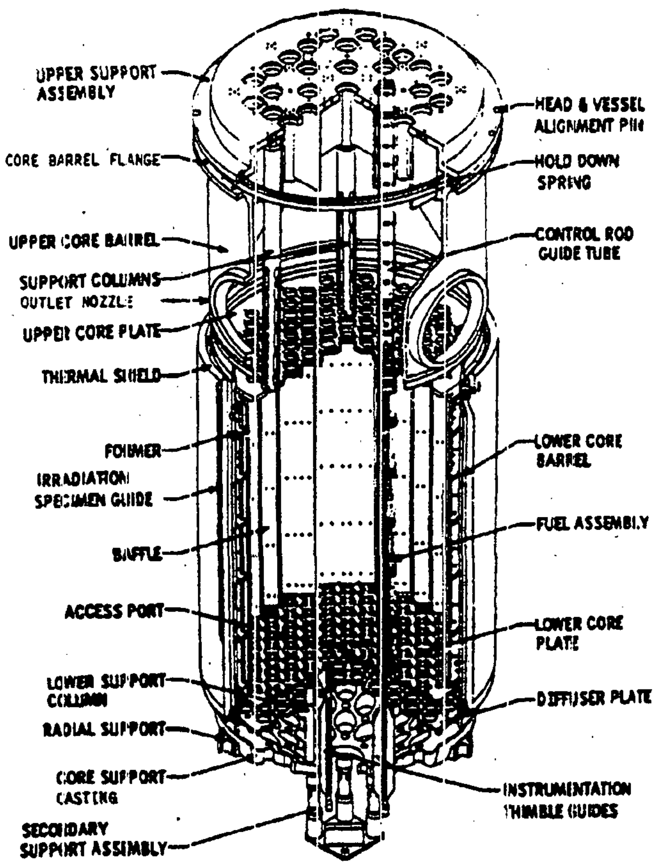
PZR

- Aging Degradation mechanisms for the IP 2 and IP3 Pressurizers are so significant that replacement is the only option for License Renewal. Some Highlights
- The pressure boundary materials of the Pressurizer are susceptible to Primary Water Stress Corrosion Cracking (PWSCC)
  - ❖ The pressurizer has Inconel 82/182 weld metal in pressurizer safety, relief, spray and surge nozzles which is susceptible cracking due to PWSCC
- The pressure boundary materials of the Pressurizer have significant end of life fatigue issues that will not meet life extension time frame
  - ❖ Fatigue of the upper portion of the pressurizer shell (44 years), the spray nozzle(49 years), the manway bolts (46 years), the seismic support lugs(41 years), lower head (due to insurge/outsurge transients), the heater wells (due to insurge and outsurge transients), the surge nozzle, the support skirt and flange (skirt –to-lower-head weld 54 years).
  - ❖ Then when you impose the NRC environmental effect to the fatigue calculations the list gets longer. Lower head (42 years), the safety and relief nozzles (53 years) and instrument nozzles (51 years)
  - ❖ This is back up by the NRC Final Safety Evaluation Report on the Acceptance for Referencing of a Generic License Renewal Program Topical Report by the Westinghouse NSSS Vendor "License Renewal Evaluation: Aging Management Evaluation For Pressurizers" dated October 26, 2000
  - ❖ Aging Management Program 2.3 needs to be imposed. This states that if the TLAA can not show acceptable usage for the license renewal period, the fatigue adequacy will be met by implementing a repair and replacement program in accordance with ASDE Section XI IWA- 004000 or IWA-7000
- NRC has issued a Final safety Evaluation Report for "Acceptance for referencing of Generic License Renewal Program Topical report entitle, "License Renewal Evaluation Aging Management Evaluation for Pressurizers" WCAP-14574 Revision 0, July 1996

Reactor Vessel Internals

- Aging Management Evaluation for Reactor Internals – WCAP –14573
- WCAP –14573 was submitted to the NRC by the Westinghouse Owners Group for IP unit 2 and Unit 3 and received a NRC Safety Evaluation Report accepting this WCAP to support License Renewal
- Reactor Vessel Intended Functions
  - Ensuring the capability to shut down the reactor and maintain it in a safe shutdown condition
  - Providing (Non – Safety Related) intended Functions that support the function listed above
  - Ensuring the integrity of the reactor coolant pressure boundary (Bottom Mounted Instrumentation Flux Thimbles Only)

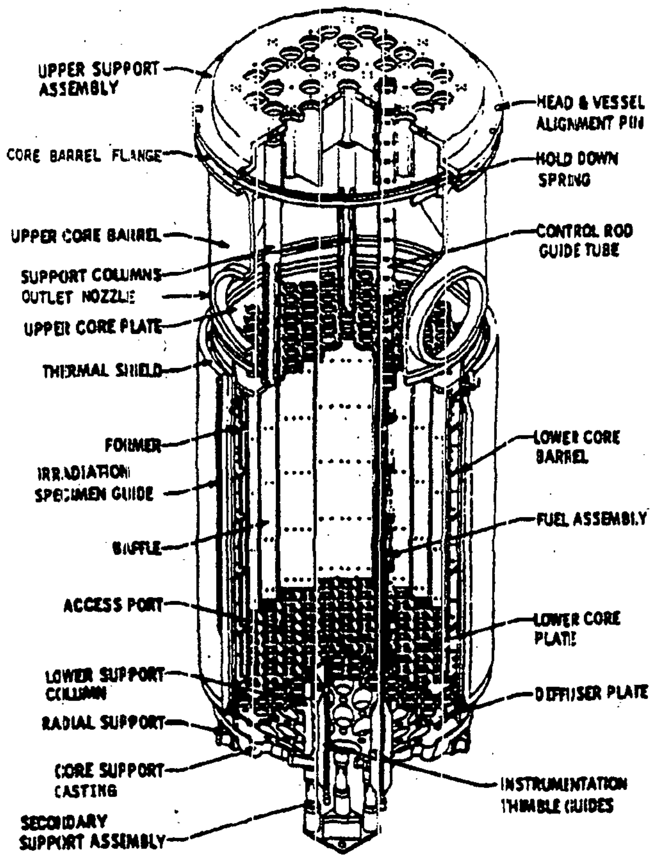
Reactor Vessel Internals



AGING MECHANISMS CONSIDERED

- IRRADIATION EMBRITTLEMENT
- STRESS CORROSION CRACKING
- IRRADIATION-ASSISTED STRESS CORROSION CRACKING
- EROSION and EROSION/CORROSION
- CREEP/IRRADIATION CREEP
- STRESS RELAXATION
- WEAR
- THERMAL AGING
- CORROSION
- FATIGUE
- SWELLING

Reactor Vessel Internals



SUMMARY OF REACTOR INTERNALS SUBCOMPONENTS REQUIRING AGING MANAGEMENT REVIEW

Part or Subcomponent	Aging Management Review Required?
Lower core plate and fuel alignment pins	YES
Lower support forging or casting	YES
Lower support columns	YES
Core barrel and core barrel flange.	YES
Radial support keys and clevis inserts	YES
Baffle and former plates	YES
Core barrel outlet nozzle	YES
Secondary core support	YES
Diffuser plate	YES
Upper support plate assembly	YES
Upper core plate and fuel alignment pin	YES
Upper support column	YES
Guide tube and flow downcomers	YES
Upper core plate alignment pin	YES
Holddown spring	YES
Head and vessel alignment pins	YES
Control rod	NO
Drive rod	YES
Neutron panels/thermal shield	YES
Irradiation specimen guide	YES
BMI columns and flux thimbles	YES
Head cooling spray nozzles	YES
Upper instrumentation column, conduit, and supports	YES
Mixing device	YES
Bolts and locking mechanisms	YES
Specimen plugs	YES

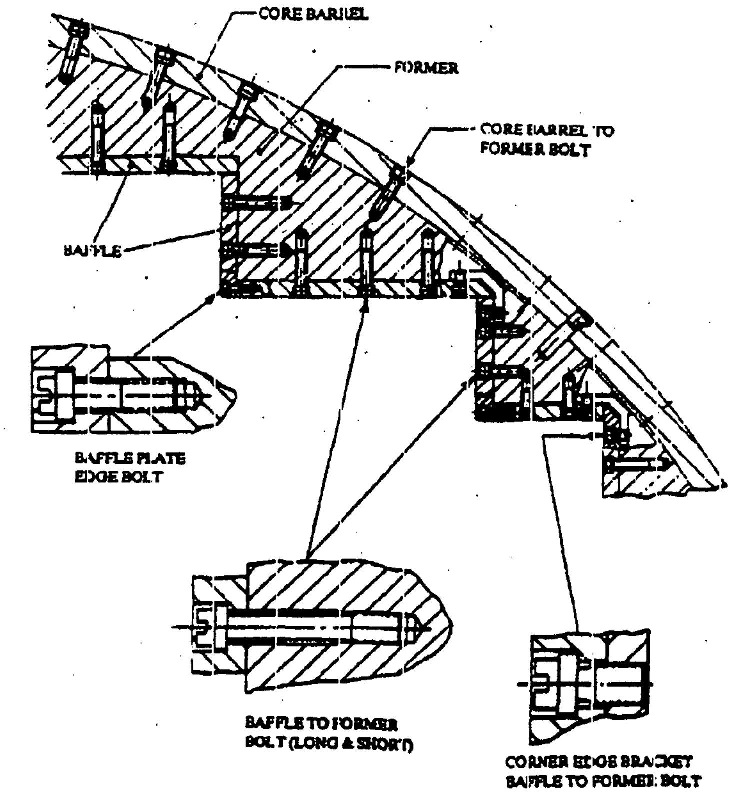
Reactor Vessel Internals

The following actions are needed for reactor vessel internals life extension as a minimum.

1. Control Rods Replacement for both Units 2 & 3
2. Specific fatigue monitoring programs for numerous Reactor Vessel Internals parts that are fatigue sensitive.
    1. Baffle Former Bolts
    2. Barrel Former Bolt
    3. Lower Core Plate
    4. Lower Support Plate
    5. Radial Key Weld
    6. Core Barrel Nozzle Weld
    7. Guide Tube/flow downcomers
    8. Upper support plate assembly

    Note these fatigue sensitive parts as calculated do not include the NRC request to include environmental effects.
3. Replacement Program for Baffle Former Bolts as a Lead Indicator for the other plant and for managing Barrel Former Bolts aging degradation. Cracked Baffle Bolts have already been replaced at Point Beach Unit 2 and RC Ginna Nuclear Power Plant in upstate New York.
4. Wear Management program for BMI flux Thimbles; Upper core plate alignment pins; radial keys and clevis inserts Per Commitments to NRC I&E Bulletin 88-09
5. Split Pin Replacement for Unit 2 with flexure modification to flexure less insert . with split pin replacement results from Unit 2, the results could be a lead indicator for Unit 3 aging management for split pins. This is only to be considered for mitigating the consequences of loose parts in the Reactor Vessel, Reactor Internals, and protection of the Steam Generators' tube sheet.

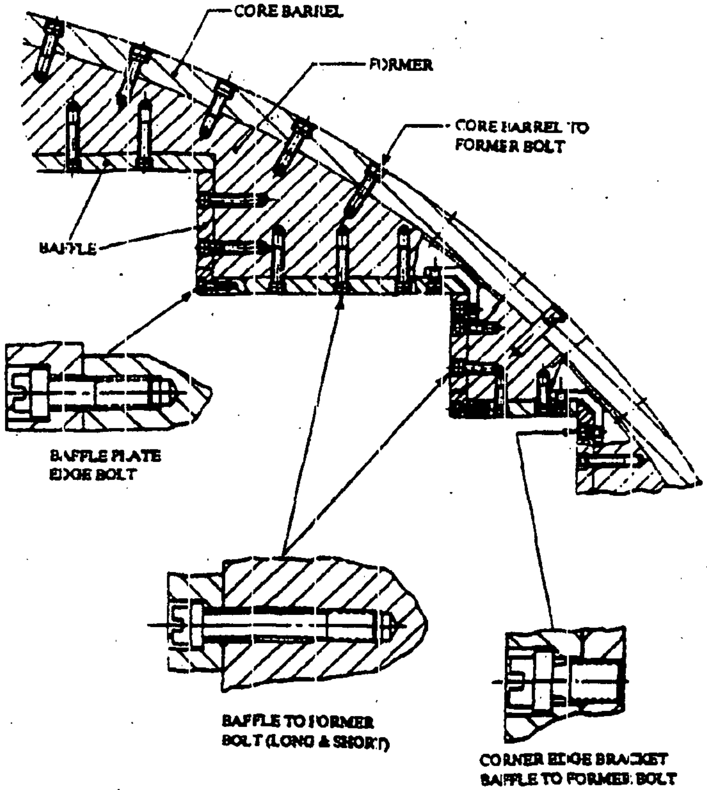
Reactor Vessel Internals



ADDITIONAL ACTIVITIES AND PROGRAM ATTRIBUTES FOR AGING MANAGEMENT OF BAFFLE/FORMER BOLTS (AMP-4.6)

Attribute	Description
Scope	Effects of cracking caused by fatigue, irradiation-induced changes in material properties, and irradiation-induced changes in stresses
Surveillance Techniques	• Visual inspection per Examination Category B-N-3 of ASME Section XI, Subsection IWB and Draft Subsection IWG • Loose parts detection monitoring system • Chemistry RC detection system • Augmented inspections (e.g., ultrasonic inspections)
Frequency	• Monitor with loose parts detection system • Monitor with RC chemistry detection system • ASME Section XI requirements, IWB-2410, -2411, -2412, -2420, -2430 and Draft IWG-2410, -2420, and -2430 • Perform sample baseline inspections prior to LR term with enhanced frequency in accordance with corrective actions
Acceptance Criteria	• Acceptable RC chemistry per technical specifications and • No loose parts from baffle/former bolt assembly and • Fatigue management program in Figure 4-1 and • Number of acceptable bolts and location ≥ the minimum number and location required to maintain core coolability and departure from nucleate boiling ratio (DNBR) within CLB limits, or if needed, for justification of continued operation (JCO), number of acceptable bolts and location ≥ JCO assumptions
Corrective Actions	The following courses of action depend on the bolt condition determined by the monitoring and inspection programs: • Supplemental examinations, analytical justifications or repair/replacement when relevant conditions are detected • Visual inspections, baffle gap measurements, augmented inspections (e.g., ultrasonic inspections), analytical justifications or repair/replacement when baffle/former bolt assembly loose parts are detected • Fuel inspections, visual baffle plate inspections, baffle gap measurements, augmented inspections (e.g., ultrasonic inspections), analytical justifications or repair/replacement when RC chemistry limits are violated • Adjustment of frequency of inspections and coverage • Analysis (e.g., fracture mechanics techniques, risk-based technology, advanced thermal/hydraulic methodologies) • Bolt replacement of a sample set so the existing bolts with indications may be analyzed (materials testing) and the new bolts monitored • Follow actions prescribed in fatigue management program
Confirmation	Acceptable performance per • Loose parts monitoring and RC chemistry programs • Augmented examinations (e.g., baffle gap inspections, ultrasonic examinations) • Analytical justification

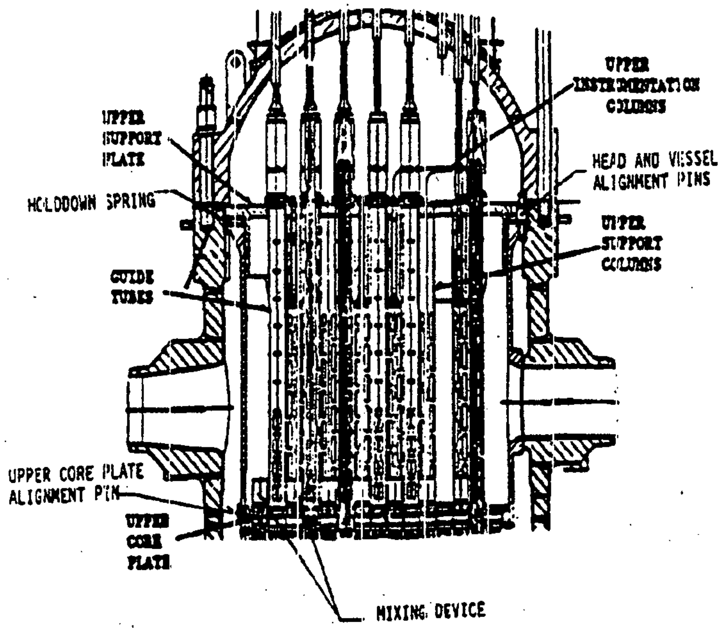
Reactor Vessel Internals



ADDITIONAL ACTIVITIES AND PROGRAM ATTRIBUTES FOR AGING MANAGEMENT OF CORE BARREL/FORMER BOLTS (AMP-4.7)

Attribute	Description
Scope	Effects of cracking caused by fatigue, irradiation-induced changes in material properties, and irradiation-induced changes in stresses
Surveillance Techniques	• Visual inspection per Examination Category B-N-3 of ASME Section XI, Subsection IWB and Draft Subsection IWG • Loose parts detection monitoring system • Augmented inspections
Frequency	• Monitor with loose parts detection system • ASME Section XI requirements, IWB-2410, -2411, -2412, -2420, -2430, and Draft IWG-2410, -2420, and -2430 • Perform sample baseline inspections prior to LR term with enhanced frequency in accordance with corrective actions
Acceptance Criteria	• No loose parts from barrel/former bolt assembly and • Fatigue management program in Figure 4-1 and • Number of acceptable bolts and location ≥ the minimum number and location required to maintain core coolability and DNBR within CLB limits, or, if needed, for JCO, number of acceptable bolts and location ≥ than JCO assumptions.
Corrective Actions	The following courses of action depend on the bolt condition determined by the monitoring and inspection programs: • Supplemental examinations, analytical justifications or repair/replacement when relevant conditions are detected • Visual inspections, augmented inspections (e.g., ultrasonic inspections), analytical justifications or repair/replacement when barrel/former bolt assembly loose parts are detected • Adjustment of frequency of inspections and coverage • Analysis (e.g., fracture mechanics techniques, risk-based technology, advanced thermal/hydraulic methodologies) • Bolt replacement of a sample set so the existing bolts with indications may be analyzed (materials testing) and the new bolts monitored • Follow actions prescribed in fatigue management program
Confirmation	Acceptable performance per • Loose parts monitoring program • Augmented examinations (e.g., ultrasonic examinations) • Analytical justification

Reactor Vessel Internals



Upper Head Aging Parts that require Aging Management Efforts

- **Guide Tubes (Guide Tubes) – Wear**
- **Control Rods – Wear, Cracking -**
- **GT Flexure Replacement for IP Unit 2- Original Flexures are susceptible to SCC**
- **Split Pins – Stress Corrosion Cracking**

Summary

- **Entergy references in the LRA their existing aging management program entitled IP3 and IP2 risk informed inservice inspection program for monitoring the welds and supports that is based on 40 operating service years not the life extension time frame 60 service years This aging management program will not adequate address life extension aging challenges to the plants pressure boundary materials and the materials in the reactor internals.**
- **The IP Unit 3 reactor vessel lower shell plate has limiting plate material does not meet the guidelines of NRC 10CFR50 Appendix G Regulations for life extension. The proposed aging management program does not assure that the reactor vessel beltline limiting plate will meet the additional 20 years of life extension.**
- **The IP3 and IP2 reactor vessel heads require a commitment to be replaced before License Renewal**
- **The pressurizer in numerous areas is fatigue sensitive and the Fatigue Cumulative Usage Factor of these numerous areas will exceed the value of 1. The aging management program proposed for monitoring this will not assure that the pressurizer's pressure boundary materials will meet the additional 20 years.**
- **The Reactor Vessel Internals has known current basis issues with fatigue and cracking of baffle bolts. Entergy has not identified in the LRA a specific aging management program that will address this situation.**
- **Scoping of Unit 1 to include Containment Structure, its spent fuel systems and support systems is justified by LRA defined intended functions scoping and screening criteria.**
- **This is just a very small sample of my evaluation (very tip of the ice berg) due to the time constraints of this meeting date. Many additional components need to be addresses e.g. Steam Generators, Reactor Coolant Pumps, Supports , RCS piping, Electrical Equipment, Containment Structures, Instrumentation, Control Rooms etc.**

EXHIBIT P

EXHIBIT Q

EXHIBIT Q

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	Docket No. 50-247
and ENTERGY NUCLEAR OPERATIONS, INC.	)	License No. DPR 26
Indian Point Entergy Center Unit 2	)	
License Renewal Application	)	

FIRST DECLARATION OF ULRICH WITTE
Review of Contentions 1 - 5

1. My name is Ulrich Witte. The "Friends United for Sustainable Energy" (FUSE) has retained me as a consultant with respect to the above-captioned proceeding. I am a mechanical engineer with over twenty-six years of professional experience in engineering, licensing, and regulatory compliance of commercial nuclear facilities. I have considerable experience and expertise in the areas of configuration management, engineering design change controls, and licensing basis reconstitution. I have authored or contributed to two EPRI documents in the areas of finite element analysis, and engineering design control optimization programs. I have led industry guidelines endorsed by the American National Standards Institute regarding configuration management programs for domestic nuclear power plants. My 26 years of experience has generally focused on assisting nuclear plant owners in reestablishing fidelity of the licensing and design bases with the current plant design configuration, and with actual plant operations. In short, my expertise is in assisting problematic plants where the regulator found reason to require the owner to reestablish competence in safely operating the facility in accordance with regulatory requirements. My curriculum vitae is attached hereto as Attachment A.

2. I submit the following comments in support of Contentions 1-5, The Applicant violated the Administrative Procedures Act in bypassing the Code of Federal Regulations (CFR) and instead used *trade* guidance for Indian Point 2 as opposed to the General Design Criteria for current design, and the current operating license with regard to the Applicant's LRA for an additional 20 years of operation .

With more than 26 years in licensing, design engineering control, configuration management. Establishing the legal ground for what Indian point 2 received its original operating license against should be straight forward in particular given the 64 design criteria that provide the fundamental framework. Essentially every other element of safety hinges on respect for the licensing and design basis, and compliance with the law, and lawful operation of the facility. One would think one could simply examine the SER, along with the rest of the CLB circa the original operating license granted and find transparent the records for design basis, construction, licensing conditions, maintenance and safe operation of the plant.

After careful examination of the facts, as represented in the table of events, it appears that just the opposite is true. Applicable rules as found in 10 CFR are not followed, and in fact it appears the applicant and the regulator are doing the opposite routinely. Bypassing the core protection provided to the public under the Administrative Procedures Act.

The past and present owners of Indian Point have failed for forty years to ensure that the nuclear reactor(s) are in compliance with regulations established by the US Nuclear Regulatory Commission to ensure public health and safety.

In its application for a 20-year license extension, Entergy has misrepresented the official record of the Federal Register to give a false appearance of compliance with regulations. In fact, the reactor has been out of compliance since it was granted its original operating license 40 years ago.

The License Renewal rule requires the applicant to identify which set of rules and regulations the reactor complies to (NRC regulations have been changed and updated several times since the 1960's.) However, the Applicant and the NRC are unable or unwilling to state which regulations are applicable to Indian Point.

The Nuclear Regulatory Commission has failed in its responsibilities by allowing Indian Point to operate under a set of "guidelines" proposed forty years ago by an industry lobbying group, but never approved by the NRC's mandatory "rule-making" process.

In its application for license extension, Entergy has failed to describe aging programs for the reactor's systems. Instead, they have "promised" compliance at some future date and time—after license renewal approval. This is a clear violation of the NRC's mandated procedures for license renewal. Indian Point 2 has serious environmental issues with spent fuel pool leaks, and radioactive leaks from underground piping that have not been addressed in their license renewal application.

The results of this are painfully obvious. A plant that experienced a design basis event tube rupture, spent fuel pools leaking, and pipes leaking. The most recent leak is cited under contention x, and was reported only a few weeks ago. The establishing of and maintaining of the design basis is impossible, when the core general design criteria are simply set aside.

Renewal of the license with the comingling of an administrative court (the NRC), and the rulemaking function of the NRC, should successfully establish the clear mandate to the licensee that laws protecting the health and safety are supposed to be followed. Not by passed.

I cannot endorse relicensing the Indian Point Unit 2 facility based upon the record and the facts of the shameful record of construction, management, and oversight of the plant.

Ulrich Witte

Witnessed and sworn to before me on this 21rst day of September, 2007

Attachment A

Ulrich K. Witte

71 Edgewood Way
Westville, Connecticut 06515
Home: 203 389 7374
Office: 860 577 8077
Mobile: 860 391 1183

Summary:

Over twenty-six years' of professional experience in engineering, configuration management, licensing, regulatory compliance of large scale commercial nuclear facilities. This includes management and implementation of design change control programs, engineering standards programs, multi-department/multi-functional licensing initiatives, plant design basis and engineering process improvement programs for six energy companies operating seven nuclear power plants. Responsibilities include:

- Systems solutions to plant operations, engineering modifications, safety analyses, design changes, installation and testing, software, drawing change programs, and training. Optimized function interfaces to insure proper coordination and synchronization for cost effective and compliant operation of the facility.
- Technical support management, and issue resolution programs that identified potential hardware, operational or equipment function issues, as well as document problems, data management problems and organizational enhancements
- Engineering Change Processes from change inception to document close-out
- Multi-department Configuration Management Program including technical approach, consensus, approval, and implementation. Managed a standing Configuration Management Programs Group whose goal was to integrate ten functional areas under a corporate strategic plan encompassing two nuclear facilities.
- Vertical slice system design/operation reviews, design bases / regulatory rule reconciliation, and licensing bases reconstitution and transitioning projects
- Integration of plant equipment information systems with business processes within engineering, materials management, maintenance, and plant operations.
- Structured business process modeling. Application of functional analysis purely from a data prospective—to enhance change management, efficiency.
- Chaired ANSI certified industry guidance on cost effective, compliant, and institutionalized programs for successful configuration management enhancement
- EPRI guidance on optimizing the Engineering Change Process
- Formal training to engineering department personal with specific courses on the engineering change process, plant safety analysis, and modification testing. Trained engineering personal on the requirements of the plant wide Configuration Management Program.

Technical Consultant

Northern Lights Engineering, L.L.C., 71 Edgewood Way, Westville, Connecticut 06515 (May 2002 –Today)

Established a consulting practice where I provided expertise in matters affecting the safe operation and regulatory compliance of commercial nuclear power facilities. This includes licensing andregulatory compliance issues, modification and implementation of industry standards, engineering design reviews, and configuration management analysis associated with an unexpected event, a design failure, or an elevated risk condition, and includes review of proposed changes to the plant operating license in preserving design efficacy.

Technical Advisor and Expert Witness to the law firm of Shems, Dunkiel, Kassel, & Saunders, PLLC
Currently providing technical assistance in prefiled testimony regarding Entergy Nuclear Operations application for renewing the operating license of Vermont Yankee. This includes Aging Review Program, in particular flow-accelerated corrosion issues, and finite element fatigue analysis reviews of susceptible components and a number of other contentions related to the safe operation of the plant beyond its 40 year license at 120% of originally design power.

Technical Advisor, to the law firm of Leroche, Meyers, and Conswel, LLP.
Provided licensing and regulatory compliance expertise in legal claim and derivative action by the board of directors of the First Energy Corporation against its corporate officers in their role associated with the Northeast black out of August 2003, and the mismanagement of the Davis Besse Nuclear Power Plant.

Technical Advisor to the Union of Concerned Scientists
Provided technical review of UCS analysis of the Davis Besse reactor head corrosion event. This included analysis of the loss of integrity of the reactor vessel, and the immediacy of the reactor head failure.

Senior Scientist, Dominion Resources Inc, Millstone Station

P.O. Box 128, Waterford, Connecticut 06385-0128 (December 1996 – 2002)

Project Manager, Licensing Commitments. Established the Regulatory Commitment Management Program. Developed a program that established senior management and department level control of more than 30,000 licensing commitment that was previously broken. The substantially enhanced program captured, dispositioned, consolidated, and managed implementation of docketed commitments to the NRC. Status, responsibility and clear communication were successfully implemented to allow Millstone to successfully restart Units 2 and 3.

The effort required substantial procedure revisions, customer consensus building, and integration of separate free-standing department specific database applications, as well as the station wide action item tracking system. A near term deliverable necessary for the successful restart of Unit 3 was to provide a workable, compliant and functioning regulatory commitment management program.

Project Manager, 50.54(f) Licensing Bases Transition Project. I led a team of 14 individuals to disposition and validate approximately 5100 regulatory commitments necessary for restart of Unit 3. The effort has led to a quality rate of more than 98 percent with production average of about four hours per commitment.

Manager, Configuration Management Program, New York Power Authority

123 Main Street, White Plains New York 10621, Nuclear Generation Department, Engineering Division
(November 1991 - November 1996)

Established the Configuration Management Program for the New York Power Authority's nuclear facilities. Included are 10 functional areas and integrated controls as authored in the corporate strategic plan. Management functions and technical skills include the following:

- Established Configuration Programs Group. This group and my position were established as a result of INPO Plant Evaluation calling for configuration management enhancement, and resolution of design control issues identified by the NRC in their DET Inspection of 1991 of the FitzPatrick Plant, as well as independent assessments. Recruited permanent staff, and supplemented the group with contracted staff on as needed basis to support both plants.

- Modified the engineering change process. Areas of immediate attention included the Design Control and Modification Programs, where a series of working groups were established to correct technical content and improve quality, ownership, and business efficiency of the design change process. This effort was achieved via: (1) a formal process to assess, model, and enhance the design change and modification process and interfaces to key functions; and (2) immediate changes to engineering procedures.

- Assessed and enhanced the Plant Equipment Data Base and controls for each plant. Results of the assessment indicated that the IP3 Plant Equipment Database contained significant problems with component classification, equipment type and status, maintenance history etc. Prepared and implemented a recovery plan and project team to reestablish the controls and content of database to be compliant with NRC Generic Letter 83-28 and to support the plant restart. Streamlined and enhanced the component classification process for both plants. Established controlled and non-controlled segregation of plant equipment in accordance with recent EPRI guidance.

- Automated and validated existing fragmented and corrupt sources of engineering information. These data sources were compiled, validated, and controlled and included multi-department areas such as set point controls, Electrical Cable and Raceway Information Systems for JAF and IP3, along with the fuse controls and data management.

- Developed design basis problem resolution process, "Design Document Open Item". Established methods for prioritizing, tracking and closing out design document issues. Established proper interface and control room notifications as per tech spec requirements. Provided guidance on operability determinations and reportability. Provided oversight for classifying and tracking more than 1100 open design issues for IP3 and 300 for JAF. Defended program to the NRC.

- Established working groups between Nuclear Generation Department and the corporate wide Information Management Organization. Gained management endorsement for areas of data quality improvement and automation for the Nuclear Generation Department. This led to enhanced implementation of the equipment information systems for both sites.

Project Manager, Program to Assure Completion and Quality, Tennessee Valley Authority:

(December 1990 - March 1991) Under contract by CYGNA Energy Services to the Vice-President, Engineering and Operations Department, Watts Bar Nuclear Plant.

- Developed a comprehensive plan to measure progress and confirm quality of the in-progress design evolution of the plant. Developed a methodology for linking specific plant equipment to that equipment's respective design basis (and associated design attributes); license commitments; and numerous verification programs currently in place. The five phase program was presented to NRR in January and received approval as an activity to assist TVA in removing the stop work order on construction of the facility.

Technical Manager, Configuration Management Program, Southern Nuclear Operating Company:

(December 1988 - November 1991). Under contract by ABB Impell and CYGNA Energy Services to Corporate Engineering Manager, Edwin I. Hatch Nuclear Plant, Georgia Power Company.

- Established and implemented the Hatch Configuration Management Program. Phase one of the effort included definition, establishment of management objectives, specification of the configuration management program scope and development of a reference manual.
- Developed and executed formal rigorous horizontal evaluations (the second phase of the project) of each relevant functional area including engineering design, implementation, plant operations and maintenance, procurement, information systems, document control and others. The program integrates functional areas across the plant, each architect engineer, and corporate (SONOPCO and Southern Company Services) organizations.
- Implemented enhancements to the program. This phase includes upgrading the design change process to achieve successful integration across organizations; stricter adherence to closure activities; and formal design engineering involvement in such activities as procurement of replacement items (equivalency). Additional controls were established such that misapplication of information obtained through informal design change processes such as the "Request for Engineering Assistance".
- Reconciling the plant's design basis. A second major activity of the program was to compile, consolidate, and ultimately, automate the plant's design basis. A major objective is to provide access and retrievability of current design basis to each of the key users of each participant organization.
- Applied Structured Business Analysis including CASE tools in the evaluation and enhancement phases. The as-found configuration management activities of all relevant processes were modeled and analyzed with this technique. Proposed enhancements are then tested on the model prior to actual implementation.
- Chaired the subcommittee for the Nuclear Information and Records Management Association which is developing a Technical Position Paper entitled, "Implementation of a Configuration Management Enhancement Program for a Nuclear Facility".

Team Leader, NRC Safety System Functional Inspection Response Organizations:

Led the NRC Safety System Functional Inspection Response Teams for Georgia Power Company (1989), and Sacramento Municipal Utility District (1987). Assisted as team coordinator in the GPC - Plant Hatch Electrical Distribution System Functional Inspection Response Team (1991). Under contract by ABB Impell (December 1987 - November 1990) to the site Engineering Manager, Rancho Seco, SMUD. and CYGNA Energy Services (December 1990 - November 1991) to the Corporate Engineering Manager, Edwin I. Hatch Nuclear Plant, Georgia Power Company.

- In the case of GPC, the NRC SSFI resulted in validation of the in progress implementation of the Hatch Configuration Management Program, and only one violation to the licensee.
- The effort included an SSFI self-assessment as well as managing the utility through the NRC inspection.
- For SMUD, developed and executed a plan for closure of both immediate findings and long term corrective action required. Assisted in defending the plan to the NRC.
- For GPC - Plant Hatch EDSFI in June 1991. Developed and implemented an EDSFI Preparation Plan for the Engineering (both A/Es) and site organizations. This effort included management of a 27 man team preparation and inspection response team for the Hatch EDSFI.

Deputy Mechanical Engineering Manager, Engineering Department

Under Contract to the Site Engineering Manager, Rancho Seco, Sacramento Municipal Utilities District, Rancho Seco (April 1986 - September 1987)

Managed the implementation and closure of over 400 modifications to the plant. Provided the NRC with a basis for allowing a successful restart of the facility. (January 1986 to November 1986) Impell Lead Project Engineer, Class 1 Piping and Support Recertification Effort, SMUD.

- Developed an engineering department action plan to improve technical quality, reconstitute design basis for five systems, control costs of plant modifications, and improve adherence to schedule.
- Responsible for the complete recertification of the Pressurizer Relief Line, Decay Heat System, and others. Responsible for expediting and implementing design changes as necessary through to closure. Assisted in Utility responses to NUREG-0737, and I&E 79-14.
- Upgraded the Engineering Department procedures to gain credit for the relaxation of ASME code requirements in structural damping values. Initiated the FSAR changes as well.

Project Engineer, Fire Protection:

Under Contract to Sacramento Municipal Utilities District, Rancho Seco (November 1984 to April 1986), SMUD Fire Protection Coordinator, Fire Protection Program

- Developed the SMUD Appendix R Fire Protection Program. Established or substantially revised 110 plant and engineering procedures including shutdown procedures on total loss of the plant's control room, technical specification surveillance procedures, fire protection system maintenance procedures, and the development of a fire protection program manual.

Successfully defended the program to the NRC during the 1985 Appendix R Inspection, with no resulting findings or open items.

Additional Experience (6/78 through 8/84):

Senior Engineer, performed original pipe stress analysis and support placement for Duke Power's Catawba Plant. Qualified approximately 8 class one and two plant systems. (ABB Impell 6/78 - 12/79).

Non-linear finite element analysis of large diameter piping for EPRI. Analysis of production stress codes versus non-linear evaluation techniques, versus actual in situ testing of the system. Results were published in EPRI Report "Seismic Piping Test and Analysis. (ABB Impell, 1980 -1981)

As Project Engineer, directed the preparation of the annual Emergency Plan exercises for Kansas Gas and Electric Company, Union Electric Company, and Texas Utilities. In two plants, the exercise was installed on the plants simulator, and received recognition from the NRC for realism of the scenario. (ABB Impell 1982-1984).

EMPLOYER SUMMARY:

Northern Lights Engineering, L.L.C. 12/2002 – current
71 Edgewood Way
Westville, CT 06515

Northeast Utilities /Dominion Resources Inc 12/1996 – 12/2002
(Under Contract via Cataract Inc through 9/97.)
2500 McClellan Ave.
Pennsauken, NJ 08109

New York Power Authority 11/1992 - 12/1996
123 Main Street
White Plains, New York 10671

Cygna Energy Services 11/1991 - 11/1992
5600 Glenridge Drive, Suite 380
Atlanta, Georgia 30075

ABB Impell Corporation 6/1978 - 11/1991
333 Research Court
Technology Park-Atlanta
Norcross, Georgia 30095

EDUCATION:

University of California, Berkeley
B.A. Physics, 1983
Senior level and graduate course work in Mechanical Engineering, and Electrical Engineering

Quinnipiac University School of Law
J.D expected June, 2009

PUBLICATIONS:

- EPRI Report Number 108736, "Guidelines for the Optimization of the Engineering Change Process," March 1994.
- NIRMA PP-03, "Position Paper for a Configuration Management Enhancement Program for a Nuclear Facility," April, 1992. Subcommittee Chair.
- EPRI Report Number 8480, " Seismic Piping Test and Analysis," 1980.

PROFESSIONAL AFFILIATIONS AND AWARDS

American Society of Mechanical Engineers, American Nuclear Society, Nuclear Information and Records Management Association, Who's Who For Rising Young Americans.

REFERENCES:

References available upon request.

EXHIBIT R

Leak found in pipe at Indian Point

By BRIAN J. HOWARD
THE JOURNAL NEWS
(Original Publication: September 7, 2007)

BUCHANAN - Workers have discovered a pinhole-sized leak in a conduit used to transfer spent fuel from the reactor to the containment pool at Indian Point 2.

The leak was found Wednesday during testing for groundwater contamination from leaks of radioactive tritium and strontium 90 that were first discovered in 2005.

"It appears that there is a potential pinhole leak in the fuel transfer canal, which we believe could be a contributing source to the groundwater contamination that we've been talking about," said Jim Steets, a spokesman for Entergy Nuclear Northeast, the plant's owner.

A vacuum test like the one that turned up the leak, as well as an ultrasonic test, will be performed to confirm the size and scope of the leak, Steets said. That will take a few more days. Repairs would follow, but would not require a reactor shutdown.

Plant officials say the leak has not contributed significantly to the groundwater contamination. The origin of the leak remains unclear.

"We'll know better about what might have caused it when we complete the testing that we're doing," Steets said. "You hate to speculate."

Nuclear Regulatory Commission spokesman Neil Sheehan said the leak was above where external moisture was found by workers during an excavation.

The leak point is under water only when the canal is flooded for refueling, which occurs every 18 to 24 months. More testing is needed before a connection can be drawn to the groundwater contamination, Sheehan said.

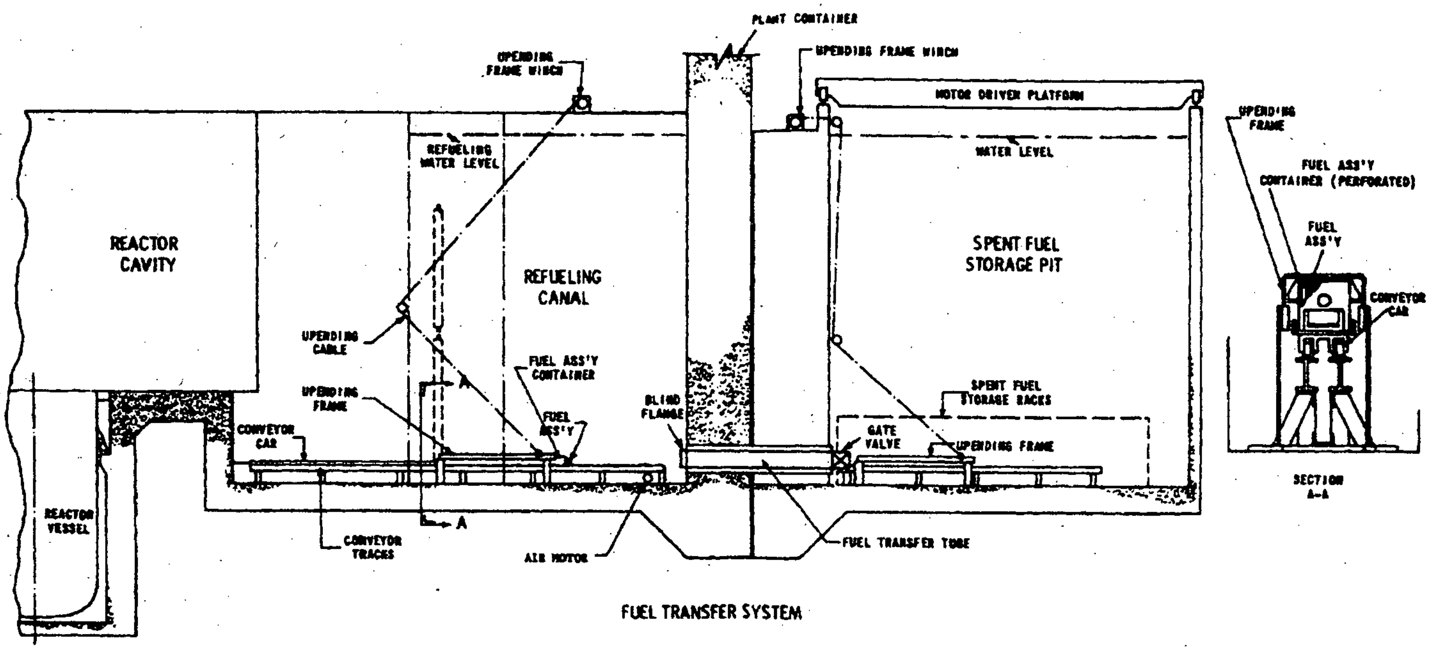
"Whether this is the cause, whether this is part of the cause, we don't know that yet, and there's still more work to be done," he said.

Buchanan Mayor Dan O'Neill learned of the leak yesterday and was assured there was no threat to the health of residents or workers at the plant.

"It does not sound like it's anything major at this time ... ." O'Neill said.

Phillip Musegaas, a staff attorney with the environmental group Riverkeeper, said the leak underscored why the NRC should require more thorough testing of systems holding radioactive water.

"This is a switch from Entergy's earlier position, because in their relicensing application they have stated that they didn't believe there was an ongoing leak at Indian Point 2 at all," Musegaas said. "The fact that they found this on further inspection suggests that they may find more leaks."



FUEL TRANSFER SYSTEM

CONSOLIDATED EDISON INDIAN POINT UNIT No. 2	
UFSAR FIGURE 9.5-1 FUEL TRANSFER SYSTEM	
MIC. No. 1999MC3886	REV. No. 16A

EXHIBIT S

Entergy

Entergy Nuclear
P.O. Box 31995
Jackson, MS 39286-1995
Tel 601 368 5692

Michael R. Kansler
President, Chief Executive Officer
& Chief Nuclear Officer

July 30, 2007
ENOC-07-0026

U.S. Nuclear Regulatory Commission
Attention: James E. Dyer
Director, Office of Nuclear Reactor Regulation
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293
Indian Point Nuclear Generating Unit No. 1
Docket No. 50-003
Indian Point Nuclear Generating Unit No. 2
Docket No. 50-247
Indian Point Nuclear Generating Unit No. 3
Docket No. 50-286
James A. FitzPatrick Nuclear Power Plant
Docket Nos. 50-333 & 72-12
Vermont Yankee Nuclear Power Station
Docket Nos. 50-271
Palisades Nuclear Plant
Docket No. 50-255 & 72-7
Big Rock Point
Docket Nos. 50-155 & 72-43

Application for Order Approving Indirect Transfer of Control of Licenses

Pursuant to Section 184 of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR 50.80, Entergy Nuclear Operations, Inc. (ENO), acting on behalf of itself and Entergy Nuclear Generation Company, Entergy Nuclear FitzPatrick, LLC, Entergy Nuclear Vermont Yankee, LLC, Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Palisades, LLC, (together, Applicants), hereby requests that the Nuclear Regulatory Commission (NRC) consent to the indirect transfer of control of the above-captioned licenses. The indirect transfer of control results from certain restructuring transactions that will involve the creation of new intermediary holding companies and/or changes in the intermediary holding companies for the ownership structure for the corporate entities that hold the NRC licenses for Pilgrim, Indian Point 1, 2, and 3, FitzPatrick, Vermont Yankee, Palisades and Big Rock Point (together, Facilities), including both the six corporate entities (named among the

A001

NMSS01

Applicants above) licensed for their ownership of the Facilities and ENO, which is the entity licensed to operate or maintain the Facilities. The licensees remain the same, and the ultimate corporate parent, Entergy Corporation, remains the same. Simplified organization charts reflecting the current and post-reorganization ownership structures are provided as Figures 1 and 2.

Through the attached Application, ENO requests, on behalf of the Applicants, that the NRC consent to this proposed indirect transfer of control. The proposed indirect transfer of control will not result in any change in the role of ENO as the licensed operator of the facilities and will not result in any changes to its technical qualifications.

In summary, the proposed indirect transfer of control will be consistent with the requirements set forth in the Act, NRC regulations, and the relevant NRC licenses and orders. The proposed indirect transfer of control will not result in any physical changes to the Facilities or changes in the officers, personnel, or day-to-day operation of the Facilities. The proposed indirect transfer of control will not involve any changes to the current licensing basis of the Facilities. It will neither have any adverse impact on the public health and safety, nor be inimical to the common defense and security. This transfer does not involve any ownership, control or domination by any foreign entity. The Applicants therefore respectfully request that the NRC consent to the indirect transfer of control of the licenses for the Facilities in accordance with 10 CFR 50.80.

ENO requests that NRC review this Application on a schedule that will permit the issuance of NRC consent to the indirect transfer of control by December 31, 2007. Such consent should be made immediately effective upon issuance and should permit the indirect transfer of control at any time for one year following NRC's approval. ENO will inform NRC if there are any significant changes in the status of any other required approvals or any other developments that have an impact on the schedule.

The Application includes a proprietary, separately bound addendum that provides Attachments 2A and 3A, which contain confidential commercial or financial information. ENO requests that Attachments 2A and 3A be withheld from public disclosure pursuant to 10 CFR 2.390, as described in the Affidavit of Michael R. Kansler, which is provided in Attachment 4 to the Application. Non-proprietary versions of Attachments 2A and 3A suitable for public disclosure are provided as Attachments 2 and 3 to the Application.

Regulatory commitments made by Entergy are identified in the table provided in the Enclosure titled "Commitments ".

If NRC requires additional information concerning this license transfer request, please contact John McCann, ENO's Director, Fleet Regulatory Affairs, at (914) 272-3370 or jmccan1@entergy.com. Service on ENO of comments, hearing requests or intervention petitions, or other pleadings, if applicable, should be made to counsel for ENO, Mr. John E. Matthews at Morgan, Lewis & Bockius, LLP, 1111 Pennsylvania Avenue, NW, Washington, DC 20004 (tel: 202-739-5524; fax: 202-739-3001; e-mail: jmatthews@morganlewis.com).

Michael R. Kansler
President & Chief Executive Officer

Enclosures: Regulatory Commitments
Oath & Affirmation
Application For Order Approving Indirect Transfer Of Control Of Licenses

cc: w/o proprietary Addendum except *

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
2443 Warrenville Rd Suite 210
Lisle, IL 60532-4352

* U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

* Steven R. Hom
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852
Mail Stop OWFN/12-D3

Mr. James J. Shea, Project Manager
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
Mail Stop 08 BI
Washington, DC 20555

Mr. John Boska, Project Manager
Plant Licensing Branch I-I
Division of Operating Reactor Licensing
Of.lce of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8-C2
Washington, DC 20555

Mahesh L. Chawla, Project Manager
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
Mail Stop 8H 4A
Washington, DC 20555

Mr. David O'Brien, Commissioner
VT Department of Public Service
112 State Street - Drawer 20
Montpelier, Vermont 15620-2601

Mr. Peter R. Smith, President
New York State Energy, Research, and
Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Eddy
New York State Department
of Public Service
3 Empire State Plaza
Albany, NY 12223

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Michigan Department of Attorney General
Special Litigation Division
525 West Ottawa St.
Sixth Floor, G. Mennen Williams Building
Lansing, MI 48913

Mr. Raymond L. Albanese
Four County Coordinator
200 Bradhurst Avenue
Unit 4 Westchester County
Hawthorne, NY 10532

USNRC Resident Inspector
Entergy Nuclear Vermont Yankee
P.O. Box 157
Vernon, Vermont 05354

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
James .A. FitzPatrick Nuclear Power Plant
PO Box 136
Lycoming, NY 13093

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Palisades Plant
27782 Blue Star Memorial Highway
Covert, MI 49043

Senior Resident Inspector's Office
Indian Point 2
U. S. Nuclear Regulatory Commission
P.O. Box 59
Buchanan, NY 10511

Senior Resident Inspector
Pilgrim Nuclear Power Station

Michigan Department of Environmental Quality
Waste and Hazardous Materials Division
Hazardous Waste and Radiological
Protection Section
Nuclear Facilities Unit
Constitution Hall, Lower-Level North
525 West Allegan Street
P.O. Box 30241
Lansing, MI 48909-7741

Mr. Robert Walker, Director
Massachusetts Department of Public Health
Schrafft Center Suite 1 M2A
Radiation Control Program
529 Main Street
Charlestown, MA 02129

Ms. Cristine McCombs, Director
Mass. Emergency Management Agency
400 Worcester Road
Framingham, MA 01702

Mr. Peter R. Smith, President
New York State Energy, Research,
& Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Supervisor
Covert Township
P. O. Box 35
Covert, MI 49043

Office of the Governor
P. O. Box 30013
Lansing, MI 48909

Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are not commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
1. For entities listed on Attachment 1 that have not yet been formed, these entities will be formed in the states indicated, with the business address indicated, and with the Directors or Managers and Executive Personnel indicated.	x		No later than the date on which the indirect license transfers are implemented.
2. Entergy Nuclear Finance Holding, LLC, will execute a financial Support Agreement in favor of the Applicants substantially in the form provided in Attachment 5.	x		No later than the date on which the indirect license transfers are implemented.
3. Entergy Nuclear Finance Holding, LLC, will provide a letter of credit or other financial assurance instrument in compliance with 10 CFR 50.75(e)(1) to be held by Entergy Nuclear Palisades, LLC and to replace the $5 million Guaranty of decommissioning funding assurance for the Big Rock ISFSI.	x		No later than the date on which the indirect license transfers are implemented.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Entergy Nuclear Operations, Inc.	)	
	)	
Pilgrim Nuclear Power Station	)	Docket Nos. 50-293
Indian Point Nuclear Generating Unit No. 1	)	50-003
Indian Point Nuclear Generating Unit No. 2	)	50-247
Indian Point Nuclear Generating Unit No. 3	)	50-286
James A. FitzPatrick Nuclear Power Plant	)	50-333 &
FitzPatrick ISFSI	)	72-12
Vermont Yankee Nuclear Power Station	)	50-271
Palisades Nuclear Plant	)	50-255 &
Palisades ISFSI	)	72-7
Big Rock Point	)	50-155 &
Big Rock Point ISFSI	)	72-043

AFFIRMATION

I, Michael R. Kansler, being duly sworn, hereby depose and state: that I am President & Chief Executive Officer, of Entergy Nuclear Operations, Inc.; that I am duly authorized to sign and file with the Nuclear Regulatory Commission the attached application for order approving indirect transfer of control of licenses; that I am familiar with the content thereof; and that the matters set forth therein are true and correct to the best of my knowledge and belief.

Michael R. Kansler
President & Chief Executive Officer

STATE OF MISSISSIPPI)
)
COUNTY OF HINDS)

Subscribed and sworn to me, a Notary Public, in and for the State of Mississippi, this 30th day of July, 2007.

Amy A. Blaylock
Notary Public in and for the State of Mississippi

Notary Public State of Mississippi At Large
My Commission Expires: June 17, 2009
Bonded Thru Heiden, Brooks & Garland, Inc.

Application for Order Approving Indirect Transfer of Control of Licenses

Entergy Nuclear Operations, Inc. (All Dockets)
Pilgrim Nuclear Power Station, Docket No. 50-293
Indian Point Nuclear Generating Unit No. 1, Docket No. 50-003
Indian Point Nuclear Generating Unit No. 2, Docket No. 50-247
Indian Point Nuclear Generating Unit No. 3, Docket No. 50-286
James A. FitzPatrick Nuclear Power Plant, Docket Nos. 50-333 & 72-12
Vermont Yankee Nuclear Power Station, Docket Nos. 50-271
Palisades Nuclear Plant, Docket No. 50-255 & 72-7
Big Rock Point, Docket Nos. 50-155 & 72-43

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TABLE OF CONTENTS

Proprietary Addendum:

I. INTRODUCTION

Pursuant to Section 184 of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR 50.80, Entergy Nuclear Operations, Inc. (ENO), acting on behalf of itself and Entergy Nuclear Generation Company, Entergy Nuclear FitzPatrick, LLC, Entergy Nuclear Vermont Yankee, LLC, Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Palisades, LLC, (together, Applicants), hereby requests that the Nuclear Regulatory Commission (NRC) consent to the indirect transfer of control of the above-captioned licenses. The indirect transfer of control results from certain restructuring transactions that will involve the creation of new intermediary holding companies and/or changes in the intermediary holding companies for the ownership structure for the corporate entities that hold the NRC licenses for Pilgrim, Indian Point 1, 2, and 3, FitzPatrick, Vermont Yankee, Palisades and Big Rock Point (together, the Facilities), including both the six corporate entities (named among the Applicants above) licensed for their ownership of the Facilities and ENO, which is the entity licensed to operate and/or maintain the Facilities. The licensees remain the same, and the ultimate corporate parent, Entergy Corporation, remains the same. Simplified organization charts reflecting the current and post-reorganization ownership structures are provided as Figures 1 and 2.

II. STATEMENT OF PURPOSE OF THE TRANSFERS AND NATURE OF THE TRANSACTION MAKING THE TRANSFERS NECESSARY OR DESIRABLE

The restructuring transactions will centralize ownership and control of the owner Applicants under a new intermediate holding company structure in the Entergy Corporation system that will be wholly owned by Entergy Nuclear Finance Holding, LLC (HoldCo.). The transactions also will centralize ownership and control of ENO and Entergy's other nuclear service businesses under Entergy Nuclear, Inc. The restructuring will enhance the financial strength of the Applicants, simplify the Applicants' and Entergy Corporation's corporate structure to the benefit of customers, regulators, capital markets and shareholders, and facilitate the financing of Holdco and its direct and indirect subsidiaries as a discrete and integrated business. The restructuring is fully consistent with the continued safe operation of the Facilities. By reorganizing a currently diffuse organization, the wholesale nuclear business will be positioned for future growth.

For historic reasons the Applicants are currently part of a dispersed structure within the Entergy Corporation system. Financing is provided internally in a top down fashion, with debt attributable to the wholesale nuclear business residing primarily with Entergy Corporation. This structure has resulted in complex financing and operating relationships. The Applicants believe that by aggregating their ownership and financing activities under Holdco within a discrete business segment structure, and aggregating their nuclear services businesses under Entergy Nuclear, Inc., they will own and operate the company's nuclear plants with more clarity and enhance their ability to attract capital.

The restructuring will create an organizational structure that is consistent with Entergy Corporation's characterization and management of the wholesale, non-utility nuclear business as one of its primary business segments. Operating revenues and net income from its nuclear services business and its wholesale, non-utility nuclear generation business will be segregated for the benefit of this business segment. This will create discrete operating history and focused operating results.

The restructuring will isolate and simplify the structure of the businesses that comprise the wholesale nuclear business segment. This simplification will enhance the ability of analysts, regulators, capital markets and shareholders to understand and evaluate this business segment.

The Applicants believe that the organization of a separate and integrated intermediate holding company system will clarify responsibilities within the Entergy Corporation system, facilitate capital formation, enhance the ability to retain and recruit qualified personnel and highlight growth opportunities for this important segment of Entergy Corporation's business.

III. GENERAL CORPORATE INFORMATION

The following are the names of the corporate entities licensed by the NRC:

Entergy Nuclear Operations, Inc.
Entergy Nuclear Generation Company
Entergy Nuclear FitzPatrick, LLC
Entergy Nuclear Vermont Yankee, LLC
Entergy Nuclear Indian Point 2, LLC
Entergy Nuclear Indian Point 3, LLC
Entergy Nuclear Palisades, LLC

The following are the names of the parent corporate entities that will directly or indirectly own the NRC licensed corporate entities.

Entergy Corporation
Entergy Nuclear, Inc.
  (by merger, successor to Entergy Nuclear Holding Company #2)
Entergy Global Trading Holdings, LTD
Entergy International Holdings, LTD
Entergy Global Investments, Inc.
  (formerly, Entergy Global, LLC)
Entergy Power Gas Holdings Corp.
Entergy Power Gas Operations Corp.
Entergy Nuclear Holding Company #1
Entergy Global Holdings, Inc.
Entergy Nuclear Finance Holding, LLC
  (formerly, Entergy Nuclear Finance Holding, Inc.)
Entergy Nuclear Holding, LLC
  (formerly, Entergy Nuclear Holding Company)
Entergy NHC, LLC
Entergy Nuclear Midwest Investment Company, LLC
Entergy Nuclear Northeast Investment Company, LLC
  (formerly, Entergy Nuclear New York Investment Company 1, and
  by merger, successor to Entergy Nuclear Holding Company #3 LLC)
Entergy Nuclear Investment Company, LLC
Entergy Nuclear Vermont Investment Company, LLC

The parent company relationships of the licensed corporate entities both before and after the indirect transfer of control are reflected in Figures 1 and 2. The information regarding each corporate entity required by 10 CFR 50.33(d)(3) is provided in Attachment 1.

All of the current and proposed directors and executive personnel of the corporate entities are citizens of the United States.

IV. FOREIGN OWNERSHIP OR CONTROL

Entergy Corporation is a publicly traded company, and its securities are traded on the New York Stock Exchange and are widely held. Section 13(d) of the Securities Exchange Act of 1934, as amended, 15 U.S.C. 78m(d), requires that a person or entity that owns or controls more than 5% of the securities of a company must file notice with the Securities and Exchange Commission (SEC). Based upon filings with the SEC, ENO is aware of one alien, foreign corporation, or foreign government that holds or may hold beneficial ownership of more than 5% of the securities of Entergy Corporation. AXA Assurance I.A.R.D. Mutuelle, a French entity, and its affiliates (together, AXA) have filed a statement indicating that as of December 31, 2006, AXA had beneficial ownership of 5% of the shares of Entergy Corporation. AXA does not have any representation on Entergy Corporation's Board of Directors, and its SEC filing specifically certifies that AXA did not acquire these shares for the purpose of or with the effect of changing or influencing the control of Entergy Corporation. See 17 CFR 240.13d-1(c)(1) (requirements for Schedule 13G filing).

The current and proposed directors and executive officers of Entergy Corporation and the Entergy subsidiaries that directly or indirectly own the Applicants are United States citizens. There is no reason to believe that the Applicants are owned, controlled, or dominated by any alien, foreign corporation, or foreign government. Thus, the indirect transfer of control of the licensed entities and their corporate parents will not result in any foreign ownership, domination, or control of these entities within the meaning of the Atomic Energy Act of 1954, as amended.

V. TECHNICAL QUALIFICATIONS

The technical qualifications of ENO are not affected by the proposed indirect transfer of control. There will be no physical changes to the Facilities and no changes in the officers, personnel, or day-to-day operations of the Facilities in connection with the indirect transfer of control. It is anticipated that ENO will at all times remain the licensed operator of the Facilities, or in the case of permanently shutdown reactors the entity licensed to maintain the Facilities.

VI. FINANCIAL QUALIFICATIONS

The Applicants are all indirect, wholly-owned subsidiaries of Entergy Corporation ("Entergy"). Headquartered in New Orleans, Louisiana, Entergy is an integrated energy company engaged primarily in electric power production and retail electric distribution operations. Entergy owns and operates power plants with approximately 30,000 MW of electric generating capacity, and Entergy is the second-largest nuclear power generator in the United States. Entergy delivers electricity to 2.6 million utility customers in Arkansas, Louisiana, Mississippi, and Texas. Entergy generated annual revenues of $10.9 billion in 2006 and had approximately 13,800 employees as of December 31, 2006. Through its subsidiaries (both regulated and non-regulated), Entergy Corporation owns and operates eleven nuclear power plants at nine sites. These include the Facilities that are the subject of this application, as well as five other nuclear power plants owned by affiliates of the Applicants: Arkansas Nuclear One Units 1 and 2, Grand Gulf Nuclear Station, River Bend Station, and Waterford 3 Steam Electric Station.

A. Projected Operating Revenues and Operating Costs

Financial information regarding Entergy Corporation and its subsidiaries is provided in its 2006 Annual Report (SEC Form 10-K) dated March 1, 2007, which is available along with Entergy's prior annual reports on the internet at:

http://www.shareholder.com/entergy/edgar.cfm?DocType=Annual,Quarterly&Year=

In addition, Applicants have prepared balance sheets and projected income statements for the licensed owners of the Facilities, as well as a projected consolidated balance sheet and projected income statement for Entergy Nuclear Finance Holding, LLC (HoldCo), which is an intermediary holding company that will indirectly own all of the corporate entities licensed to own the Facilities, as well as other assets and businesses related to non-utility nuclear generation business of Entergy Corporation.

ENO, the corporate entity licensed to operate the operating Facilities and to maintain the non-operating Facilities, will be a wholly-owned subsidiary of Entergy Nuclear, Inc., which itself will be a direct wholly-owned subsidiary of Entergy Corporation. Entergy Nuclear, Inc, will own the nuclear services businesses of Entergy Corporation. ENO will receive the revenue necessary to operate and maintain the Facilities, including decommissioning funds to pay for such expenses, from the corporate entities licensed to own the Facilities pursuant to operating agreements or other intra-corporate arrangements that have been previously described to NRC. If any changes are made to replace the existing arrangements, any new agreements are expected to be consistent with the current arrangements. Any new agreements will be made available for inspection by NRC. As such, ENO relies upon the financial qualifications of the licensed owners of the Facilities, because these corporate entities will be financially responsible for the operation and decommissioning of the units.

In accordance with 10 CFR 50.33(f) and the Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance (NUREG-1577, Rev. 1) ("Standard Review Plan"), projected balance sheets for each of the licensed owners of the Facilities are provided in a separately bound proprietary addendum as Attachment 2A. In addition, a projected opening balance sheet for the consolidated businesses of Entergy Nuclear Finance Holding, LLC is also provided in Attachment 2A. ENO requests that Attachment 2A be withheld from public disclosure, as described in the Affidavit provided in Attachment 4. Redacted versions of these balance sheets, suitable for public disclosure, are provided as Attachment 2.

In addition, *pro forma* Projected Income Statements for the six year period from January 1, 2007 through December 31, 2012 for each of the licensed owners of the Facilities and Entergy Nuclear Finance Holding, LLC are provided in a separately bound proprietary addendum as Attachment 3A. In addition, a sensitivity analysis of these projections (reflecting a 10% reduction in projected revenue) is provided in Attachment 3A. ENO requests that Attachment 3A be withheld from public disclosure, as described in the Affidavits provided in Attachment 4. Redacted versions of these balance sheets, suitable for public disclosure, are provided as Attachment 3.

The Projected Income Statements for the licensed owners show that anticipated revenues from sales of capacity and energy from the Facilities provide reasonable assurance of an adequate source of funds to meet the ongoing operating and maintenance expenses for the Facilities. In addition, Entergy Nuclear Finance Holding, LLC will execute a financial Support Agreement with the Applicants, including each of the corporate entities licensed to own the Facilities, in the total amount of $700 million, to pay for the operating and maintenance (O&M)

costs for all six operating Facilities, if called upon to do so. This provides assurance that adequate funds will be available to fund ongoing O&M expenses with respect to all of the operating Facilities. A form of this agreement is provided as Attachment 5.

The financial projections for Entergy Nuclear Finance Holding, LLC establish that it will have adequate resources from its consolidated businesses to provide funding if necessary under the Support Agreement. In addition, this parent company is expected to have access to a line of credit of at least $1 billion or more, which provides additional assurance of its ability on an ongoing basis to provide funds for the licensed entities.

Pursuant to the Support Agreement, the licensed owners will have access to funds sufficient to pay the fixed O&M costs in the event of any unanticipated plant shutdown in accordance with the guidance provided in the Standard Review Plan. Pursuant to this agreement, Entergy Nuclear Finance Holding, LLC will make up to an aggregate amount of $700 million in funding available to any and all of the Applicants to meet their obligations to NRC relating to the Facilities. This arrangement replaces the prior financial support arrangements under which funds were available to each licensed owner individually in limited amounts, and Applicants seek NRC's prior written approval of the revocation of the prior arrangements through NRC's approval of the new Support Agreement, which rescinds the prior arrangements under the terms of Section 7 of the Support Agreement.

Under the new Support Agreement, each of the licensed entities will have access to up to a total of $700 million, to the extent not previously utilized, for any single plant outage or for a multiple plant outage should the circumstances necessitate access to such funds. As such, the proposed Support Agreement would provide funding for any individual site that significantly exceeds the six-month period suggested by the NRC's Standard Review Plan guidance, which requests demonstration of a source of funds to pay fixed O&M expenses in the event of an extended plant outage. The availability of the entire aggregate amount of funding under the Support Agreement for each plant is superior to the current disparate support arrangements. Moreover, the total amount available would fund nearly six-months worth of fixed O&M expenses for all six operating Facilities. Finally, Applicants note that they do not expect to need to request funding under this formal agreement, as they expect that during their day-to-day operations and otherwise as the need for funding arises, they will have access to funds from capital contributions, loans, credit lines, or other sources that provide adequate funding to support safe operation of all of the Facilities.

B. Decommissioning Funding

The financial qualifications of the Applicants to continue to own the Facilities are further demonstrated by the decommissioning funding assurance provided in accordance with 10 CFR 50.75(e)(1). Details regarding the status of the decommissioning funding assurance maintained by the Applicants for the Facilities are provided in the March 29, 2007 decommissioning funding status report (ENOC-07-00007) submitted by ENO in accordance with 10 CFR 50.75(f), except for Palisades and Big Rock Point which were not included in this report. This report demonstrates that there is reasonable assurance of adequate decommissioning funding that is provided by pre-paid amounts maintained as assets in external sinking funds segregated from licensee assets and outside licensee administrative control in accordance with the requirements of 10 CFR 50.75(e)(1)(i).

With respect to Palisades, the trust fund balance for Palisades as of April 30, 2007 was approximately $252.9 million, and with credit for earnings taken into account as permitted by NRC rules, less than $205 million in pre-paid assets maintained in a trust would be sufficient to fully fund the NRC's current "formula amount" estimate for Palisades decommissioning costs at

$345.9 million, calculated pursuant to 10 CFR 50.75(c). Thus, the existing trust fund balances maintained by Entergy Palisades LLC as assets in an external sinking fund segregated from licensee assets and outside licensee administrative control provide decommissioning funding assurance in accordance with the requirements of 10 CFR 50.75(e)(1)(i). There is, therefore, reasonable assurance that the amount of decommissioning funds available will be sufficient to pay decommissioning costs for Palisades at the time permanent termination of operations is expected.

With respect to Big Rock Point, the NRC acknowledged in its recent approval of the transfer of this facility to Entergy Palisades LLC that NRC has approved the release of most of the Big Rock Point site, and the remaining decommissioning obligation is approximately $2.8 million estimated for the decommissioning of the Independent Spent Fuel Storage Facility (ISFSI). Entergy Corporation committed to provide a Parent Guaranty for $5 million. Prior to the indirect transfer of the Big Rock Point license, this Parent Guaranty will be terminated and replaced by an alternative financial assurance mechanism acceptable under the terms of 10 CFR 50.75(e)(1), such as a letter of credit from a financial institution or a pre-paid decommissioning trust in an amount not less than $2.8 million. None of the other existing arrangements for Big Rock Point as approved in the prior license transfer will be affected. This provides reasonable assurance of the availability of funds for decommissioning the Big Rock Point ISFSI pursuant to 10 CFR 50.75 and 72.30.

Other than the changes to the Parent Guaranty for Big Rock Point described above, the Applicants do not anticipate any changes in the existing decommissioning funding assurance provided in connection with the proposed indirect transfers of control. Applicants also do not anticipate any changes or amendments to any nuclear decommissioning trust fund agreements, and if any amendments are to be made in the future, the existing trust agreements require prior written notice be provided to the NRC. Moreover, any existing NRC license conditions governing these trust agreements will remain in effect and unchanged.

VII. ANTITRUST INFORMATION

This Application post-dates the issuance of the operating licenses of the facilities, and therefore no antitrust review is required or authorized. Based upon the Commission's decision in *Kansas Gas and Electric Co., et al.* (Wolf Creek Generating Station, Unit 1), CLI-99-19, 49 NRC 441 (1999), the Atomic Energy Act of 1954, as amended, does not require or authorize antitrust reviews of post-operating license transfer applications.

VIII. RESTRICTED DATA AND CLASSIFIED NATIONAL SECURITY INFORMATION

The proposed transfers do not involve any Restricted Data or other Classified National Security Information or result in any change in access to such Restricted Data or Classified National Security Information. ENO's existing restrictions on access to Restricted Data and Classified National Security Information are unaffected by the proposed transfers. In compliance with Section 145(a) of the Act, the applicants agree that restricted or classified defense information will not be provided to any individual until the Office of Personnel Management investigates and reports to the NRC on the character, associations, and loyalty of such individual, and the NRC determines that permitting such person to have access to Restricted Data will not endanger the common defense and security of the United States.

IX. ENVIRONMENTAL CONSIDERATIONS

The requested consent to indirect transfer of control of the facilities' licenses is exempt from environmental review because it falls within the categorical exclusion contained in 10 CFR 51.22(c)(21), for which neither an Environmental Assessment nor an Environmental Impact

Statement is required. Moreover, the proposed indirect transfer does not involve any amendment to the facility operating licenses or other change, and it will not directly affect the actual operation of the Facilities in any substantive way. The proposed transfer does not involve an increase in the amounts, or a change in the types, of any radiological effluents that may be allowed to be released off-site, and involves no increase in the amounts or change in the types of non-radiological effluents that may be released off-site. Further, there is no increase in the individual or cumulative operational radiation exposure, and the proposed transfer has no environmental impact.

X. PRICE-ANDERSON INDEMNITY AND NUCLEAR INSURANCE

The proposed indirect transfer of control does not affect the existing Price-Anderson indemnity agreements for the Facilities, and does not affect the required nuclear property damage insurance pursuant to 10 CFR 50.54(w) and nuclear energy liability insurance pursuant to Section 170 of the Act and 10 CFR Part 140.

XI. EFFECTIVE DATE AND OTHER REQUIRED REGULATORY APPROVALS

Accordingly, ENO requests that NRC review this Application on a schedule that will permit the issuance of NRC consent to the indirect transfer of control by December 31, 2007. Such consent should be made immediately effective upon issuance and should permit the indirect transfer of control at any time within a year after issuance. ENO will inform the NRC if there are any significant changes in the status of any other required approvals or any other developments that have an impact on the schedule.

XII. CONCLUSION

Based upon the foregoing information, ENO respectfully requests, on behalf of the Applicants, that the NRC issue an Order consenting to the indirect transfer of control.

FIGURE 1

SIMPLIFIED ORGANIZATION CHART – CURRENT

Figure 1: SIMPLIFIED ORGANIZATION CHART – CURRENT

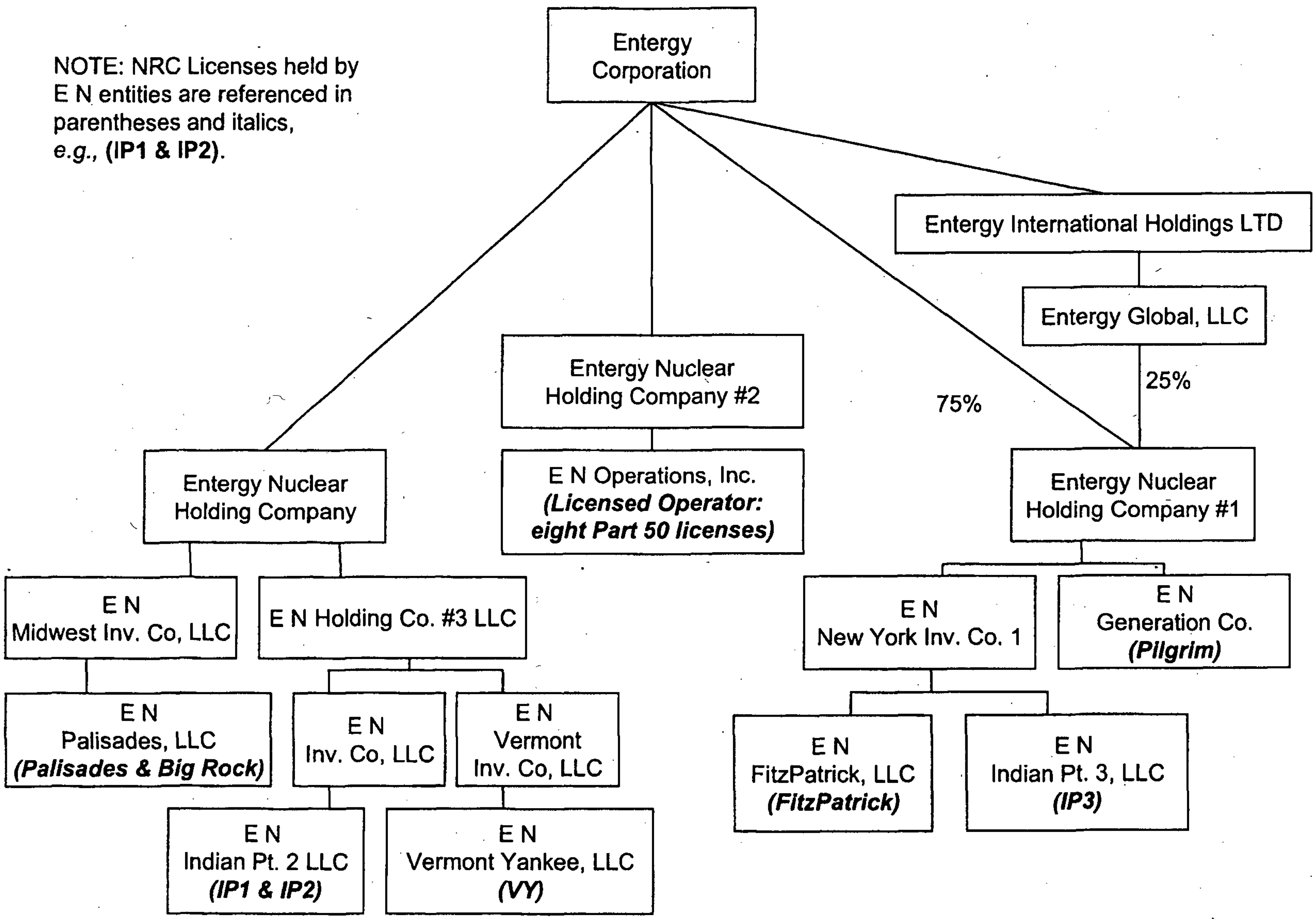
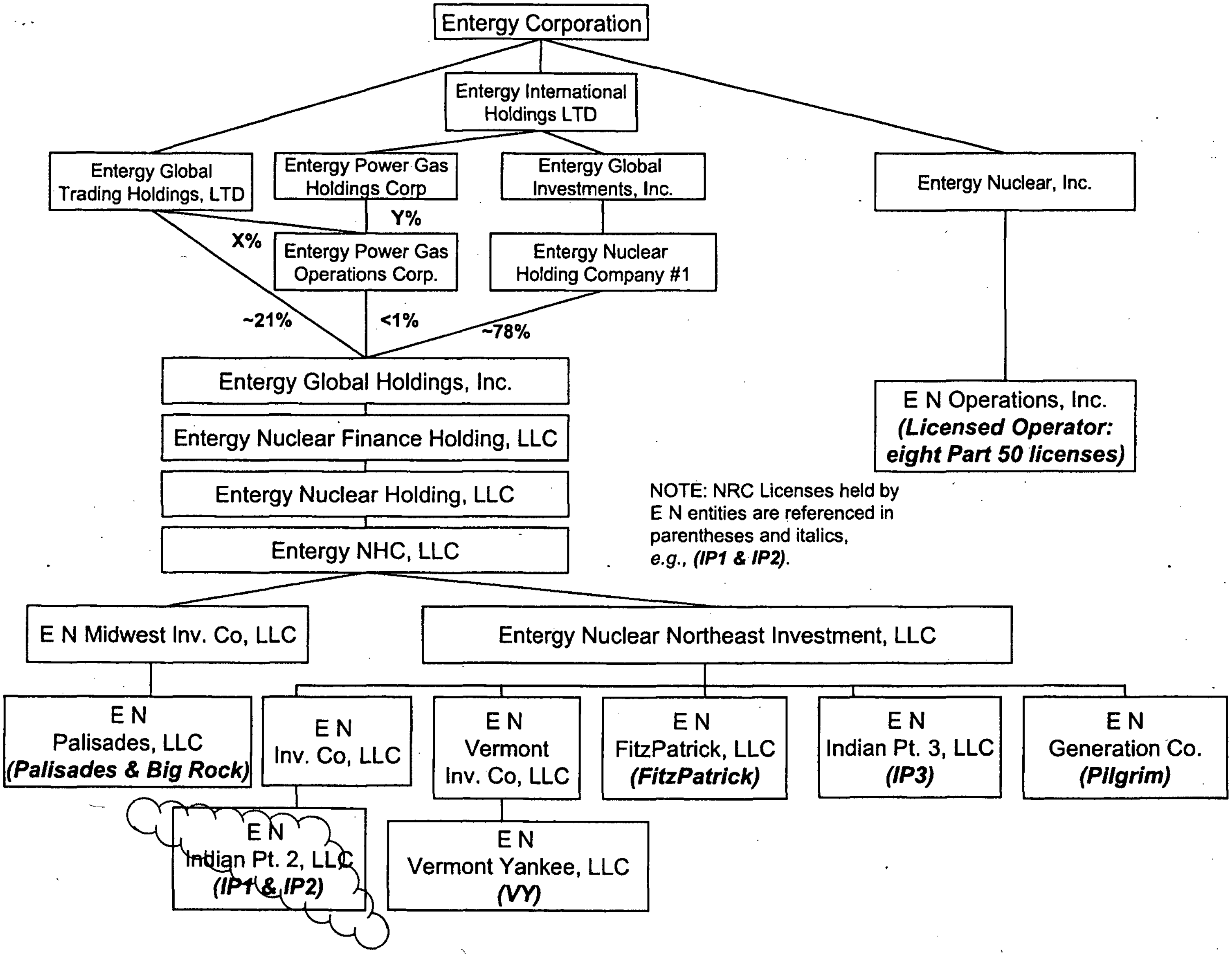


FIGURE 2

SIMPLIFIED ORGANIZATION CHART – POST REORGANIZATION

Figure 2: SIMPLIFIED ORGANIZATION CHART – POST REORGANIZATION



ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Corporation
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	639 Loyola Avenue New Orleans, LA 70113
DIRECTORS:	J. Wayne Leonard (Chairman) Maureen S. Bateman W. Frank Blount Simon D. deBree Gary W. Edwards Alexis M. Herman Donald C. Hintz Stuart L. Levinick James R. Nichols William A. Percy, II W. J. "Billy" Tauzin Steven V. Wilkinson
EXECUTIVE PERSONNEL	J. Wayne Leonard – Chief Executive Officer Richard J. Smith – President & Chief Operating Officer Gary J. Taylor – Group President, Utility Operations Leo P. Denault – Executive VP & CFO Curtis L. Hebert, Jr. – Executive VP, External Affairs Michael R. Kansler – Executive VP & Chief Nuclear Officer Mark T. Savoff – Executive VP, Operations Robert D. Sloan – Executive VP / General Counsel & Secretary Theodore H. Bunting, Jr – Senior VP & Chief Accounting Officer Joseph T. Henderson – Senior VP & General Tax Counsel Terry R. Seamons – Senior VP, Human Resources & Administration Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear, Inc.
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
DIRECTORS:	Michael R. Kansler – Chairman Leo P. Denault C. Randy Hutchinson
EXECUTIVE PERSONNEL	Michael R. Kansler – President & Chief Executive Officer C. Randy Hutchinson – Senior VP, Development Robert D. Sloan – Executive VP & Secretary Wanda Curry – VP Chief Financial Officer, Nuclear Operations Danny R. Keuter – VP Business Development Steven C. McNeal – VP & Treasurer Dana Atchison – Assistant Secretary Amy A. Blaylock – Assistant Secretary Terence A. Burke – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Global Trading Holdings, LTD
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	10055 Grogans Mill Road, Parkwood II Building The Woodlands, TX 77380
DIRECTORS:	Barrett E. Green John Wengler James E. Striedel
EXECUTIVE PERSONNEL	Barrett E. Green – President John Wengler – VP & Treasurer James E. Striedel – Vice President Thomas Wagner – Secretary Joseph T. Henderson – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy International Holdings LTD
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	20 Greenway Plaza, Suite 500 Houston, TX 77046
DIRECTORS:	Steven C. McNeal Eddie Peebles Andrew Rosenlieb
EXECUTIVE PERSONNEL	Eddie Peebles – President Steven C. McNeal – Vice President & Treasurer Andrew Rosenlieb – Vice President & Secretary Thomas G. Wagner – Assistant Secretary Joseph T. Henderson – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Global Investments, Inc. (Proposed Conversion)
STATE OF INCORPORATION:	Arkansas
BUSINESS ADDRESS:	425 West Capitol Avenue Little Rock, AR 72201
DIRECTORS:	Douglas Castleberry Steven C. McNeal O. H. Storey, III
EXECUTIVE PERSONNEL	Douglas Castleberry – President Robert D. Sloan – Executive VP, General Counsel, & Secretary Steven C. McNeal – Vice President & Treasurer O. H. Storey, III – Vice President Sue Chambers – Assistant Secretary Janan E. Honeysuckle – Assistant Secretary Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Power Gas Holdings Corporation
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	20 Greenway Plaza, Suite 500 Houston, Texas 77046
DIRECTORS:	Steven C. McNeal
EXECUTIVE PERSONNEL	James E. Striedel – President Joseph T. Henderson – Tax Officer Steven C. McNeal – VP & Treasurer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Power Gas Operations Corporation
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	Entity Services (Nevada), L.L.C. 2215-B Renaissance Dr., Suite 5 Las Vegas Nevada 89119
DIRECTORS:	Richard F. Boland Douglas Castleberry Steven C. McNeal Tom D. Reagan
EXECUTIVE PERSONNEL	Tom D. Reagan – President Richard F. Boland – VP, Secretary, & Assistant Treasurer Steven C. McNeal – VP & Treasurer Thomas G. Wagner – Assistant Secretary Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Holding Company #1
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	20 Greenway Plaza, Suite 500 Houston, Texas 77046
DIRECTORS:	Michael R. Kansler (Chairman) Wanda Curry
EXECUTIVE PERSONNEL	Michael R. Kansler – President & Chief Executive Officer Joseph T. Henderson – Senior VP & General Tax Counsel Wanda Curry – VP Thomas G. Wagner – Secretary Paul A. Castanon – Assistant Secretary Rory L. Roberts – Tax Officer Steven C. McNeal – VP & Treasurer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Global Holdings, Inc. (Proposed Entity/Not Yet Created)
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	10055 Grogans Mill Road, Parkwood II Building The Woodlands, TX 77380
DIRECTORS:	James E. Striedel* Andrew Rosenlieb*
EXECUTIVE PERSONNEL	James E. Striedel* – President Andrew Rosenlieb* – Vice President John Wengler* – VP & Treasurer Reginald G. Rice* – Secretary Joseph C. Henderson* – Tax Officer

*Subject to additional internal review by Affiliate Rules Compliance

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Finance Holding, LLC (Proposed Conversion)
STATE OF INCORPORATION:	Arkansas
BUSINESS ADDRESS:	425 West Capitol Little Rock, AR 72201
MANAGERS:	Douglas Castleberry – Management Committee Member Michael R. Kansler – Management Committee Member O. H. Storey – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – President & Chief Executive Officer Douglas Castleberry – Vice President Steven C. McNeal – VP & Treasurer O. H. Storey – VP & Secretary Sue Chambers – Assistant Secretary Janan E. Honeysuckle – Assistant Secretary Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Holding, LLC (Proposed Conversion)
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	20 Greenway Plaza, Suite 500 Houston, Texas 77046
MANAGERS:	Wanda Curry – Management Committee Member Eddie Peebles – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – President & Chief Executive Officer Robert D. Sloan – Executive VP & Secretary Joseph T. Henderson – Senior VP & General Tax Counsel Wanda Curry – Vice President Steven C. McNeal – VP & Treasurer Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy NHC, LLC (Proposed Entity/Not Yet Created)
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	10055 Grogans Mill Road, Parkwood II Building The Woodlands, TX 77380
MANAGERS:	James E. Striedel* Andrew Rosenlieb*
EXECUTIVE PERSONNEL	James E. Striedel* – President Andrew Rosenlieb* – Vice President John Wengler* – VP & Treasurer Reginald G. Rice* – Secretary Joseph C. Henderson* – Tax Officer

* Subject to additional internal review by Affiliate Rules Compliance

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Midwest Investment Company, LLC
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
MANAGERS:	C. Randy Hutchinson – Management Committee Member
EXECUTIVE PERSONNEL	Joseph T. Henderson – Senior VP & General Tax Counsel Terence A. Burke – VP & Secretary Steven C. McNeal – VP & Treasurer Amy A. Blaylock – Assistant Secretary Paul A. Castanon – Assistant Secretary David Gibbs – Assistant Secretary Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Northeast Investment Company, LLC (Proposed Conversion)
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
DIRECTORS OR MANAGERS:	Michael R. Kansler – Management Committee Member C. Randy Hutchinson – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – President, Executive VP & Chief Executive Officer Terence A. Burke – VP & Secretary Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Investment Company, LLC
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
MANAGERS:	C. Randy Hutchinson – Management Committee Member Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Terence A. Burke – VP & Secretary Amy A. Blaylock – Assistant Secretary Paul A. Castanon – Assistant Secretary Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Vermont Investment Company, LLC
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
MANAGERS:	C. Randy Hutchinson – Management Committee Member Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Terence A. Burke – VP & Secretary Paul A. Castanon – Assistant Secretary Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Operations, Inc. [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
DIRECTORS:	C. Randy Hutchinson Michael R. Kansler
EXECUTIVE PERSONNEL	Michael R. Kansler –Chief Executive Officer John McGaha – President, Planning, Development & Oversight John T. Herron – Senior VP, Entergy Nuclear Operations C. Randy Hutchinson – Senior VP, Business Development Robert D. Sloan – Executive VP, General Counsel & Secretary Michael A. Balduzzi, Jr. – Senior VP, Chief Operating Officer, ENO Kevin Bronson – VP Operations, Pilgrim Wanda Curry – VP, Chief Financial Officer, Nuclear Fred R. Dacimo – VP Operations, Indian Point Energy Center Peter T. Dietrich – VP Operations, JAF Danny R. Keuter – VP, Development, Planning & Innovation Oscar Limpias – VP, Engineering Steven C. McNeal – VP & Treasurer Stewart B. Minahan – VP Operations, Cooper Christopher J. Schwarz – VP Operations, Palisades Theodore A. Sullivan – VP Operations, Vermont Yankee Amy A. Blaylock – Assistant Secretary Terence A. Burke – Assistant Secretary Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer Paul Hinnenkamp – VP, Business Development Cliff Eubanks – VP, Project Management Joseph DeRoy – VP, Operations Support Bruce Williams – VP, Oversight

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Generation Company [NRC Licensed Entity]
STATE OF INCORPORATION:	Massachusetts
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
DIRECTORS:	Michael R. Kansler – Chairman C. Randy Hutchinson
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President Robert D. Sloan – Executive VP & Secretary John T. Herron – Senior VP & Chief Operating Officer Michael A. Balduzzi, Jr. – VP, Operations, Pilgrim NPS Wanda Curry – VP, Chief Financial Officer, Nuclear Terence A. Burke – VP & Secretary Steven C. McNeal – VP & Treasurer Amy A. Blaylock – Assistant Secretary Paul A. Castanon – Assistant Secretary James W. Snider – Assistant Secretary Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear FitzPatrick, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	268 Lake Road East Lycoming, New York 13093
MANAGERS:	Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President John T. Herron – Senior VP & Chief Operating Officer Robert D. Sloan – Executive VP, General Counsel & Secretary Wanda Curry – VP, Chief Financial Officer, Nuclear Peter T. Dietrich – VP, Operations Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Vermont Yankee, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	320 Governor Hunt Road Vernon, Vermont 05302
MANAGERS:	Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President Robert D. Sloan – Executive VP, General Counsel & Secretary John T. Herron – Senior VP & Chief Operating Officer Wanda Curry – Vice President, Chief Financial Officer, Nuclear Operations Steven C. McNeal – VP & Treasurer Theodore A. Sullivan – VP, Operations Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Indian Point 2, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	Bleakley Avenue and Broadway Buchanan, New York 10511
MANAGERS:	Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President Robert D. Sloan – Executive VP & Secretary John T. Herron – Senior VP & Chief Operating Officer Wanda Curry – VP, Chief Financial Officer, Nuclear Operations Fred R. Dacimo – Vice President, Operations Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Indian Point 3, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	Bleakley Avenue and Broadway Buchanan, New York 10511
MANAGERS:	Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President John T. Herron – Senior VP & Chief Operating Officer Robert D. Sloan – Executive VP, General Counsel & Secretary Wanda Curry – VP, Chief Financial Officer, Nuclear Operations Fred R. Dacimo – Vice President, Operations Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Palisades, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	27780 Blue Star Memorial Highway Covert, Michigan 49043
MEMBER (MEMBER MANAGED LLC):	Entergy Nuclear Midwest Investment Company, LLC – Member
EXECUTIVE PERSONNEL	Michael R. Kansler – President Terence A. Burke – VP & Secretary Steven C. McNeal – VP & Treasurer Christopher J. Schwartz – VP, Operations Dana Atchison – Assistant Secretary Amy A. Blaylock – Assistant Secretary Paul A. Castanon – Assistant Secretary David Gibbs – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

ATTACHMENT 2

Projected Balance Sheets: 2007-2012

(Non-Proprietary Version)

Entergy Nuclear Finance Holding, LLC (Consolidated) -- Projected Balance Sheets (2007-2012)

Dollars in Thousands

Forecast as of April 2007

	Projected Balance as of December 31					
	2007	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability and Other						
Notes Payable (1)						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity (1)						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 2, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands	**Projected Balance as of December 31**					
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 3, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands

Forecast as of April 2007

	Projected Balance as of December 31					
	2007	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant (1)						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Vermont Yankee, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands	Projected Balance as of December 31					
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear FitzPatrick, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands	Projected Balance as of December 31					
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant (1)						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Generation Company -- Projected Balance Sheets (2007-2012)

Dollars in Thousands			Projected Balance as of December 31			
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Palisades, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands			**Projected Balance as of December 31**			
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability and Other						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

ATTACHMENT 3

Projected Income Statements: 2007-2012

(Non-Proprietary Version)

Entergy Nuclear Finance Holding, LLC (Consolidated) -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Entergy Nuclear MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 2, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
Indian Point 2 MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
IP-1						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						

Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 3, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
IP3 MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						

Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Vermont Yankee, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Vermont Yankee MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income (1)						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Vermont Yankee, LLC -- Cash Flow Statements (2007-2012)
Dollars in Thousands
Forecast as of April 2007

	2007	2008	2009	2010	2011	2012
OPERATING ACTIVITIES						
Net Income						
Non-Cash Items Included in Net Income:						
Depreciation, Amortization, Decommissioning and Deferred Income Taxes						
Other						
NET CASH FLOW PROVIDED BY (USED IN) OPERATING ACTIVITIES						
INVESTING ACTIVITIES						
Construction Expenditures						
Nuclear Fuel Purchase						
Decommissioning Trust Contributions and Realized Changes in Trust Assets						
NET CASH FLOW PROVIDED BY (USED IN) INVESTING ACTIVITIES						
FINANCING ACTIVITIES						
Proceeds from Issuance of:						
Long-Term Debt						
Retirement of:						
Long-Term Debt						
Notes from Parents / Associated Companies						
Other						
NET CASH FLOW PROVIDED BY (USED IN) FINANCING ACTIVITIES						
Net Increase (Decrease) in Cash and Cash Equivalents						
Cash and Cash Equivalents at Beginning of Period						
CASH AND CASH EQUIVALENTS AT END OF PERIOD						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear FitzPatrick, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Fitzpatrick MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Generation Company -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007

	2007	2008	2009	2010	2011	2012
Pilgrim MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Palisades, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
Palisades MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Big Rock ISFI						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Finance Holding, LLC (Consolidated) -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)
Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Entergy Nuclear MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						

Note: Assumes 01/01/08 Close

Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 2, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Indian Point 2 MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
IP-1						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECASE STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 3, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)
Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
IP3 MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						

Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

Entergy Nuclear Vermont Yankee, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)
Dollars in Thousands

Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
Vermont Yankee MDC	605	605	605	605	605	605
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear FitzPatrick, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)

Dollars in Thousands

Forecast as of April 2007

	2007	2008	2009	2010	2011	2012
Fitzpatrick MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						

Note: Assumes 01/01/08 Close

Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Generation Company -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction In Revenue)
Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Pilgrim MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						

Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

Entergy Nuclear Palisades, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)
Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Palisades MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Big Rock ISFI						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						

Note: Assumes 01/01/08 Close

Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Attachment 4

10 CFR 2.390 AFFIDAVIT OF MICHAEL R. KANSLER

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Entergy Nuclear Operations, Inc.	)	
	)	
Pilgrim Nuclear Power Station	)	Docket Nos. 50-293
Indian Point Nuclear Generating Unit No. 1	)	50-003
Indian Point Nuclear Generating Unit No. 2	)	50-247
Indian Point Nuclear Generating Unit No. 3	)	50-286
James A. FitzPatrick Nuclear Power Plant	)	50-333
Vermont Yankee Nuclear Power Station	)	50-271
Palisades Nuclear Plant	)	50-255

AFFIDAVIT

I, Michael R. Kansler, President & Chief Executive Officer of Entergy Nuclear Operations, Inc. (ENO), hereby affirm and state:

1. I am authorized to execute this affidavit on behalf of ENO.

2. ENO is providing information in support of an Application for Order Approving Indirect Transfer of Control of Licenses. The documents being provided in Attachment 2A and 3A contain proprietary financial information and financial projections related to the ownership and operation of the generation assets operated by ENO. These documents constitute proprietary commercial and financial information that should be held in confidence by the NRC pursuant to 10 CFR § 2.390(a)(4) because:

    i. This information is and has been held in confidence by ENO.

    ii. This information is of a type that is customarily held in confidence by ENO and there is a rational basis for doing so because the information contains sensitive financial information concerning projected revenues and operating expenses of ENO.

    iii. This information is being transmitted to the NRC voluntarily and in confidence.

    iv. This information is not available in public sources and could not be gathered readily from other publicly available information.

    v. Public disclosure of this information would create substantial harm to the competitive position of ENO by disclosing its internal financial projections.

3. Accordingly, ENO requests that the designated documents be withheld from public disclosure pursuant to 10 CFR § 2.390(a)(4).

Michael R. Kansler

STATE OF MISSISSIPPI	)
	)
COUNTY OF HIN~~E~~DS	)

Subscribed and sworn to me, a Notary Public, in and for the State of Mississippi, this 30th day of July, 2007.

Amy A. Blaylock

Notary Public in and for the
State of Mississippi

Notary Public State of Mississippi At Large
My Commission Expires: June 17, 2009
Bonded Thru Heiden, Brooks & Garland, Inc.

Attachment 5

Form of SUPPORT AGREEMENT

Between

Entergy Nuclear Finance Holding, LLC

and

Entergy Nuclear Generation Company,
Entergy Nuclear FitzPatrick, LLC,
Entergy Nuclear Vermont Yankee, LLC,
Entergy Nuclear Indian Point 2, LLC,
Entergy Nuclear Indian Point 3, LLC, and
Entergy Nuclear Palisades, LLC

THIS SUPPORT AGREEMENT, dated as of ___________ __, 2007 between Entergy Nuclear Finance Holding, LLC, a Delaware corporation ("Parent"), and Entergy Nuclear Generation Company, Entergy Nuclear FitzPatrick, LLC, Entergy Nuclear Vermont Yankee, LLC, Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Palisades, LLC (individually, "Subsidiary Licensee" and collectively, "Subsidiary Licensees"),

WITNESSETH:

WHEREAS, through its intermediate subsidiary companies, Parent is the indirect owner of 100% of the outstanding shares of the Subsidiary Licensees;

WHEREAS, the Subsidiary Licensees are the corporate entities that hold the NRC licenses for Pilgrim, Indian Point 2 and 3, FitzPatrick, Vermont Yankee, and Palisades (individually, each a "Facility," and collectively the "Facilities"); and

WHEREAS, Parent and the Subsidiary Licensees desire to take certain actions to assure the ability of the Subsidiary Licensees to pay the pro rata expenses of maintaining the Facilities safely and protecting the public health and safety (the "Operating Expenses") and to meet Nuclear Regulatory Commission ("NRC") requirements during the life of each Facility (the "NRC Requirements").

NOW, THEREFORE, in consideration of the mutual promises herein contained, the parties hereto agree as follows:

1. *Availability of Funding.* From time to time, upon request of Subsidiary Licensees, Parent shall provide or cause to be provided to Subsidiary Licensees such funds as the Subsidiary Licensees determine to be necessary to pay Operating Expenses and meet NRC Requirements; provided, however, in any event the aggregate unreimbursed amount which Parent is obligated to provide under this Agreement at any one time shall not exceed $700 million.

2. *No Guarantee.* This Support Agreement is not, and nothing herein contained, and no action taken pursuant hereto by Parent shall be construed as, or deemed to constitute, a direct or indirect guarantee by Parent to any person of the payment of the Operating Expenses or of any liability or obligation of any kind or character whatsoever of the Subsidiary Licensees. This Agreement may, however, be relied upon by the NRC in determining the financial qualifications of each Subsidiary Licensee to hold the operating license for a Facility.

3. *Waivers.* Parent hereby waives any failure or delay on the part of the Subsidiary Licensees in asserting or enforcing any of their rights or in making any claims or demands hereunder.

4. *Amendments and Termination.* This Agreement may not be amended or modified at any time without 30 days prior written notice to the NRC. This Agreement shall terminate at such time as Parent is no longer the direct or indirect owner of any of the shares or other ownership interests in a Subsidiary Licensee. This Agreement shall also terminate with respect to the Operating Expenses and NRC Requirements applicable to a Facility whenever such Facility permanently ceases commercial operations and certification is made as to the permanent removal of fuel from the reactor vessel.

5. *Successors.* This Agreement shall be binding upon the parties hereto and their respective successors and assigns.

6. *Third Parties.* Except as expressly provided in Sections 2 and 4 with respect to the NRC, this Agreement is not intended for the benefit of any person other than the parties hereto, and shall not confer or be deemed to confer upon any other such person any benefits, rights, or remedies hereunder.

7. *Other Financial Support Arrangements.* This Agreement supersedes any other support arrangement relating to NRC requirements, if any exists prior to the date hereof, between Parent or any other affiliate that is a signatory hereto, and a Subsidiary Licensee to provide funding when necessary to pay Operating

Expenses and meet NRC Requirements for the Facilities, and any such other financial support arrangement is hereby voided, revoked and rescinded. As such, the total available funding provided for in this Agreement shall be limited as set forth in Section 1 herein and shall not be cumulative with any other financial support arrangement for purposes of meeting NRC requirements that is subject to the jurisdiction of the NRC. For avoidance of doubt, the parties agree that this Section 7 does not apply to financial guarantees or commitments made to third parties, even where such agreements may relate to compliance with NRC requirements. A list of the financial support arrangements rescinded by this paragraph is provided as Schedule A.

8. *Governing Law.* This Agreement shall be governed by the laws of the State of Delaware.

IN WITNESS WHEREOF, the parties hereto have caused this Agreement to be executed and delivered by their respective officers thereunto duly authorized as of the day and year first above written.

ACKNOWLEDGED AND AGREED

Entergy Nuclear Finance Holding, LLC

By:
Name:________________________
Title:________________________

Entergy Corporation

By:
Name:________________________
Title:________________________

Entergy International Holdings LTD

By:
Name:______________________________
Title:______________________________

Entergy International LTD LLC

By:
Name:______________________________
Title:______________________________

Entergy Nuclear Generation Company

By:
Name:______________________________
Title:______________________________

Entergy Nuclear FitzPatrick, LLC

By:
Name:______________________________
Title:______________________________

Entergy Nuclear Vermont Yankee, LLC

By:
Name:______________________________
Title:______________________________

Entergy Nuclear Indian Point 2, LLC

By:
Name:______________________________
Title:______________________________

Entergy Nuclear Indian Point 3, LLC

By:
Name:____________________________
Title:_____________________________

Entergy Nuclear Palisades, LLC

By:
Name:____________________________
Title:_____________________________

Schedule A

Guarantor	Guaranty on behalf of	Amount	Claim
Entergy International LTD LLC	Entergy Nuclear Generation Company	$50M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy International LTD LLC	Entergy Nuclear Indian Point 2, LLC	$35M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy International Holdings LLC	Entergy Nuclear Vermont Yankee, LLC	$35M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy Corporation	Entergy Nuclear Vermont Yankee, LLC	$35M	If the financial assurance line is below $35M at the point that it is determined that ENVY will cease operations, ETR will make additional funds available to ENVY, up to $35M.
Entergy Corporation	Entergy Nuclear Vermont Yankee, LLC	$25M	If the financial assurance line is below $25M at the point that it is determined that ENVY will cease operations, ETR will make additional funds available to ENVY, up to $25M.
Entergy International LTD LLC	Entergy Nuclear FitzPatrick, LLC & Entergy Nuclear Indian Point 3, LLC	$50M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy Global, LLC	Entergy Nuclear FitzPatrick, LLC	$20M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy Global, LLC	Entergy Nuclear Indian Point 3, LLC	$20M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy Global, LLC	Entergy Nuclear Palisades, LLC	$25M	Guarantee to NRC for financial assurance to provide for safe plant operation through working credit line.

~~EXHIBIT T~~

Exhibit V

Synapse
Energy Economics, Inc.

FINANCIAL INSECURITY: The Increasing Use of Limited Liability Companies and Multi-Tiered Holding Companies to Own Nuclear Power Plants

Prepared by:
David Schlissel, Paul Peterson and Bruce Biewald
Synapse Energy Economics
22 Pearl Street, Cambridge, MA 02139
www.synapse-energy.com
617-661-3248

Prepared for:
STAR Foundation
Riverkeeper, Inc.

August 7, 2002

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List of Tables and Attachments

Foreword

Where Have All the Safeguards Gone

In nuclear power's first two decades, accident insurance requirements were seriously inadequate. Decommissioning costs were overlooked entirely. The 1979 accident at Three Mile Island undermined much nuclear complacency. In the early 1980s Congress and the U.S. Nuclear Regulatory Commission made serious efforts to address these shortcomings.

The nuclear self-insurance requirement – known as the Price-Anderson Act – was increased from $560 million to the current $9.3 billion, and each plant was required to set up a dedicated decommissioning trust fund to assure that funds would be available to clean up a closed plant.

With the passage of two more decades, renewed complacency has eroded these safeguards.

This report dissects a troublesome set of developments on the cusp between economic and safety regulation, namely the rearrangement of nuclear power plant ownership into the limited liability subsidiaries of a few large companies. Because this arrangement has occurred during an era of lax and dispirited regulation, some important issues have not been pursued effectively. As a result, the consolidation of nuclear ownership – although probably a positive development if carried out wisely – now risks the shifting of accident and decommissioning costs from the plant owners to the general public because the relatively secure financial backing of substantial utility companies has in many cases been replaced by a limited liability subsidiary whose only asset is an individual nuclear power plant.

With years of reckless undermining of economic and financial regulation now exposed in a series of catastrophic financial collapses, investigators turning over rocks keep finding the same agents of decay: demands for short term "performance" in the private sector compounded by regulatory cutbacks, underqualified commission appointments, Congressional hearings harassing public protection initiatives, pressure to deregulate more and faster-a ruinous mixture of money, pressure, overconfidence, complexity and ideology.

During all those years, health and safety regulation got the same debilitating treatment from Congress and the Presidency as its financial counterparts. How long before those chickens come home to roost, and where will the roosting be?

Even in the best of times, regulation tends to be reactive, responding to events or to applications. Rarely does a regulatory commission develop a set of affirmative requirements to guide those who seek its permits. Certainly neither the Nuclear Regulatory Commission nor the several economic regulators with jurisdiction over nuclear plants ever developed a comprehensive policy to guide those seeking to transfer nuclear plant ownership. Such a policy might have required a showing that the protection of the public was in no way diminished by these transfers. Or such a requirement might have been imposed as a condition of approving the transfers.

But it was not.

In the absence of any such requirement, public protection has depended on the acumen of a Nuclear Regulatory Commission unversed in financial matters and of economic regulators unversed in health and safety issues. As has happened in financial and in utility restructuring circles, fundamental safeguards have been circumvented.

Regulating in this way is like driving drunk. On any one occasion, there may be no consequences at all. But in the nuclear field the possibilities include the undermining of the scheme that assures compensation in the event of nuclear accidents and an increased likelihood that some of the costs of decommissioning nuclear power plants will be borne by the general public. Taxpayers, utility customers and powerplant neighbors who thought themselves protected by firm requirements may one day wear the stunned expressions of Enron retirement plan holders or WorldCom investors.

Clever advisors in several professions have no doubt been well rewarded for achieving these "deregulations." As they were at Global Crossing. As they were at Qwest. As they were at Andersen Consulting. But in the nuclear realm as in the others, they have been more clever than wise. The consequences remain to be revealed. We will be fortunate if the only harm is another blow to public confidence.

Peter Bradford[1]

[1] Visiting Lecturer in Energy Policy and Environmental Protection, Yale University; Former Chair, New York State Public Service Commission and Maine Public Utilities Commission; Former Commissioner, U.S. Nuclear Regulatory Commission; Past President, National Association of Regulatory Utility Commissioners.

Introduction

In recent years corporations have increasingly owned and operated nuclear power plants through multiple tiered holding companies, which frequently include limited liability companies ("LLCs"). LLCs are new organizational forms whose liability is limited to the specific assets they directly own. More than 25 nuclear power plants are today owned by such LLCs and additional corporate reorganizations can be expected. The use of complex organizational structures involving LLCs can shield the parent corporations and their shareholders from liabilities incurred by both direct and indirect subsidiaries. In so doing, the use of multi-tiered holding companies and LLCs to own and operate nuclear power plants raises several concerns regarding security, safety and potential federal and consumer liabilities.

Nuclear power plants were traditionally constructed and operated mainly by integrated investor-owned utilities under "cost-of-service regulation" through which necessary funds were provided to operate and decommission the plants safely. Starting in the mid-1990s, however, many nuclear power plant owners began to reorganize and to sell their nuclear units to unaffiliated companies or corporate affiliates. Some of these corporate reorganizations were required or encouraged as part of state efforts to deregulate the electric utility industry and to implement industry restructuring. Other reorganizations were adopted by plant owners, on their own initiative, in order to minimize tax liabilities, maximize flexibility in corporate ownership and management, and to protect corporate assets. According to the U.S. General Accounting Office ("GAO"), the U.S. Nuclear Regulatory Commission ("NRC") has reviewed more than 60 license transfer requests in recent years, affecting more than half of the nuclear plants in the nation.[2]

Synapse Energy Economics, Inc. ("Synapse") was asked by the STAR Foundation and Riverkeeper, Inc. to survey the increasing use of complex corporate ownership structures and LLCs to own and operate nuclear power plants and to review the NRC's oversight of these developments. Synapse also was asked to identify those areas in which changes need to be made to assure that there are adequate funds available to meet NRC-imposed requirements, including post September 11, 2001 security-related requirements and Price-Anderson Act nuclear accident insurance obligations and to assure that decommissioning funds are adequate and are protected. This Report presents our findings.

Data Sources

Synapse has used publicly available documents from the following sources in the preparation of this Report: the U.S. GAO, the U.S. NRC, corporate filings at the U.S. Securities and Exchange Commission, company websites, nuclear industry publications, utility filings at state regulatory commissions and answers to post-hearing questions that arose out of the January 23, 2002 Price-Anderson Act Hearings. The specific documents on which this Report is based are identified in footnotes or the list of references.

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[2] *Nuclear Regulation: NRC's Assurance of Decommissioning Funding During Utility Restructuring Could be Improved,* U.S. GAO Report, GAO-02-48, December 2001, at page 21.

This Report also relies on detailed publicly available information about the Entergy Corporation that Synapse obtained as a result of its work in Vermont Public Service Board Docket No. 6545 in which Entergy's proposed acquisition of the Vermont Yankee nuclear plant has been examined.

Conclusion

Over the last ten years, the ownership of an increasing number of nuclear power plants has been transferred to a relatively small number of very large corporations. These large corporations have adopted business structures that create separate limited liability subsidiaries for each nuclear plant, and in a number of instances, separate operating and ownership entities that provide additional liability buffers between the nuclear plant and its ultimate owners. The limited liability structures being utilized are effective mechanisms for transferring profits to the parent/owner while avoiding tax payments. They also provide a financial shield for the parent/owner if an accident, equipment failure, safety upgrade, or unusual maintenance need at one particular plant creates a large, unanticipated cost. The parent/owner can walk away, by declaring bankruptcy for that separate entity, without jeopardizing its other nuclear and non-nuclear investments. This report examines the recent trend towards the use of limited liability corporations in the nuclear industry, often as part of multi-tiered holding companies, and identifies numerous concerns related to the use of such business structures.

Summary of Findings

The above conclusion is based on the following findings:

Finding No. 1 - Nuclear power plant ownership and operation has become increasingly consolidated in a small number of very large corporations..

Finding No. 2 – Complex, holding companies, often including Limited Liability subsidiaries, are increasingly being used to own nuclear power plants.

Finding No. 3 – Limited Liability Companies are relatively new business structures that can enhance a parent corporation's ability to transfer funds from its subsidiaries and to shield assets from liability for financial risks.

Finding No. 4 –There continue to be significant financial and other risks associated with nuclear power plant ownership and operations.

Finding No. 5 – The NRC has expressed concern that deregulation can adversely affect the safety of operating nuclear power plants by increasing the pressure on licensees to reduce costs.

Finding No. 6 – The NRC has expressed concern that the use of holding company structures can reduce the assets that would be available for the safe operation and decommissioning of a nuclear power plant. However, the NRC does not adequately protect against the risk that an LLC subsidiary will transfer all of its operating profits to its parent company or engage in risky loans to or questionable deals with affiliated companies.

Finding No. 7 - The NRC's reviews of the financial qualifications of new nuclear power plant owners are inconsistent and may be too limited to ensure that subsidiaries will have adequate funds to safely operate and decommission their nuclear plants and pay retrospective Price-Anderson Act premiums.

Finding No. 8 – The financial guarantees that the NRC requires from prospective nuclear power plant owners may not be adequate to assure that plants are operated and decommissioned safely and that plant owners will be able to pay retrospective Price-Anderson Act insurance premiums in the event of a nuclear accident.

Finding No. 9 - The NRC has proposed to significantly reduce its review of a non-electric utility licensee's financial qualifications when it evaluates an application to renew a nuclear plant's operating license.

Finding No. 10 – The NRC does not require that parent corporations guarantee that funds will be provided to safely operate and decommission the nuclear power plants owned by their subsidiary companies.

Finding No. 11 – Taxpayers may be at risk if nuclear plant owning subsidiaries are unable to continue making safety-related or decommissioning expenditures or pay retrospective Price-Anderson Act premiums.

Finding No. 12 – The NRC has no statutory authority to require a licensee in bankruptcy to continue making safety-related or decommissioning expenditures or to pay retrospective Price-Anderson Act premiums.

Finding No. 13 – Case law indicates that it could be very difficult to hold a parent corporation responsible for the liabilities incurred by nuclear power plant-owning LLC subsidiaries in a multi-tiered holding company.

Finding No. 14 – The NRC has expressed serious doubts as to its ability to hold a parent corporation responsible for the liabilities incurred by a subsidiary.

Finding No. 15 – Shielding parent corporations from nuclear power plant operating, accident insurance, and decommissioning risks is unfair and economically inefficient.

Recommendations

1. Parent corporations should be required to guarantee that plant-owning subsidiaries and affiliates will be provided whatever funds are needed to safely operate and decommission their nuclear power plants.

2. Parent corporations should be held fully responsible for the unmet liabilities incurred by both direct and indirect nuclear power plant owning subsidiaries.

3. Congress should adopt legislation to assure that costs related to (1) safety and security (2) decommissioning assets and (3) Price-Anderson nuclear accident responsibilities receive priority in bankruptcy proceedings.

4. Reactor owners should be required to guarantee payment of their nuclear accident insurance responsibilities under the Price-Anderson Act through surety bonds,

letters of credit, sinking funds, or other comparable financial instruments that will be bankruptcy remote. This will assure that public liability claims will be paid up to the limits of the Price-Anderson Act without concern about the financial condition of the industry and without requiring a taxpayer bailout.

5. The Nuclear Regulatory Commission should not eliminate the current legal requirement that non-utility corporations must disclose their financial qualifications when applying to re-license nuclear power plants, as the agency has proposed in a recent rulemaking. Instead, the NRC should bolster its disclosure requirements concerning the character of the legal relationships between a parent corporation and its subsidiaries in the event of a bankruptcy, business failure or accident.

Finding No. 1 - Nuclear power plant ownership and operation has become increasingly consolidated in a small number of very large corporations.

In the past, a relatively large number of utilities around the nation owned nuclear power plants or, at least, were joint owners with other companies. However, as a result of industry restructuring, nuclear power plant ownership has become increasingly consolidated in a small number of large corporations. In fact, as shown in Table No. 1 below, ten corporations currently own all or part of 70 of the 103 nuclear power plants in the U.S.

Table No. 1
Concentration of Nuclear Power Plant Ownership

Parent Corporation	Number of Operating Nuclear Units Owned (in whole or in part)
Exelon Corporation	19
Entergy Corporation	10[3]
Duke Energy	6
Dominion Resources, Inc.	6
Southern Company	6
TVA	5
Progress Energy	5
FPL Group	5[4]

[3] Includes the Vermont Yankee Nuclear Station.

[4] Includes the Seabrook Nuclear Station.

Constellation Energy Group, Inc.	4
FirstEnergy	4

The six largest owners alone own part or all of 52 nuclear units, or one-half of all of the operating nuclear power plants in the nation.

At the same time, the Nuclear Management Company ("NMC") holds the NRC-issued operating licenses for eight nuclear plants in the Midwest. However, each of the utilities involved in NMC continues to own its own plants, is entitled to the energy generated by the plants, and retains the financial obligations for the plants safe operation, maintenance and decommissioning.

This industry consolidation may yield significant benefits in terms of economics, safety and reliability. However, it also raises the possibility that simultaneous extended outages of more than one nuclear plant will leave a "fleet" owner without needed revenues to fund safety-related expenses or capital expenditures at its other facilities. At the same time, the increasing consolidation of ownership also raises the possibility that an owner will have to bear Price-Anderson Act retrospective burdens measured in the hundreds, not tens, of millions of dollars, possibly without adequate revenues from which to make such payments.[5]

In fact, there have been numerous instances where two or more of a company's nuclear plants have been out of service at the same time for six months or longer due to problems that arose as a result of an emphasis on reducing costs, deficiencies in the utility's safety culture, management problems, or generic or plant-specific technical issues. For example:

- Two of the three units at the Palo Verde Nuclear Generating Station were shut down at the same time for approximately twelve months starting in March 1989. During this same twelve month period, the third Palo Verde unit was shut down for numerous outages, including one outage that lasted approximately four months.
- The two units at the South Texas nuclear plant were both shut down for the twelve month period February 1993 to February 1994.
- All five of TVA's operating nuclear power plants were shut down in 1985. The first unit to be restarted, Sequoyah Unit 1, re-commenced commercial operations in May 1989.
- Northeast Utilities' Millstone Units 2 and 3 were shut down for multi-year outages between March 1996 and June 1998. Millstone Unit 1 was shutdown in November 1995 and permanently retired in 1997.

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5 In the event of an accident at one of the nation's nuclear power plants, the Price-Anderson Act requires nuclear plant owners to make "retrospective" (i.e., post-accident) payments after the initial $200 million tier of insurance is exhausted. Under the Act's present terms, and given the number of operating plants, this obligation is a maximum of $88.085 million per unit with a maximum of $10 million per year. Consequently, an owner of multiple units could face retrospective obligations of hundreds of millions of dollars in total and tens of millions per year.

- Commonwealth Edison experienced numerous simultaneous extended outages among the eight units at its Dresden, LaSalle, Quad Cities, and Zion nuclear stations. For example, during the first six months of 1996, the utility had at least three units shut down at any one time for extended outages of longer than three months in duration. Commonwealth Edison had at least four units shut down at any one time for extended outages during the last six months of 1996, except for a short period at the end of August and early September. The utility also experienced simultaneous outages of at least six months in length at its two unit Zion nuclear station from October 1993 through April 1994 and at its two unit LaSalle Station from September 1996 through 1998.
- Both units at the D.C. Cook Nuclear Plant in Michigan were shutdown from September 1997 through June 2000.
- Both units at the Salem Nuclear Station were shutdown for more than two years between July 1995 and August 1997.
- Both units at the Brunswick nuclear plant were shutdown for the twelve month period April 1992 through April 1993.
- Both units at the Calvert Cliffs nuclear plant were shut down at the same time for more than one year starting in May 1989.

Finding No. 2 - Complex, multi-tiered holding companies, often including limited liability subsidiaries, are increasingly being used to own nuclear power plants.

Except for those power plants owned by municipal utilities and the Yankee Nuclear Plants in the Northeast, nuclear units historically were directly owned by integrated investor-owned utility companies which owned other generating facilities and had significant transmission and distribution assets as well. Over the past five to ten years, however, corporations have established multiple tiered holding companies through which they indirectly own nuclear power plants. Except for the Exelon Corporation, these new nuclear power plant owning subsidiaries generally own only a single asset, i.e., an individual nuclear power plant, or both units at a multiple unit site.[6]

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[6] The nuclear industry's interest in single asset nuclear generating companies is not new. It dates back to the 1960s, perhaps even to the 1950s, when the plans were developed for the ownership of the Yankee Rowe and Connecticut Yankee nuclear plants. Then, in the late 1980s and early 1990s, some companies, including Middle South Utilities (subsequently renamed "Entergy") and General Public Utilities, reorganized, creating specific corporate entities to operate ***but not own*** their nuclear power plants. In one notable case in Michigan, however, the Consumers Power Company proposed transferring a poorly performing nuclear plant, Palisades, to a new corporate entity, PGCo, created for the sole purpose of owning and operating the plant. This ill-conceived proposal was designed to shift nuclear-related risks away from the Company, placing them instead upon consumers and the public. For more information, see Bruce Biewald, "Do We Really Need Nuclear Generating Companies?," in Public Utilities Fortnightly, June 7, 1990. and the Direct Testimony of Bruce Biewald, submitted on behalf of the Attorney General of Michigan, April 19, 1989 in Michigan Public Service Commission Case No. U-9172.

The corporate subsidiaries included in these complex ownership chains are increasingly chartered as Limited Liability Companies ("LLCs"). As we will discuss in Finding No. 3 below, LLCs are relatively new business structures that enhance a parent corporation's ability both to transfer funds from its nuclear-power plant owning subsidiaries and to shield its other assets from liability from the financial risks associated with its nuclear operations.

The following examples illustrate the accelerating trend in the nuclear industry to use multiple tiered holding companies and LLC subsidiaries to own and operate nuclear plants. It is important to note that each of the parent corporations listed in these examples also has numerous other subsidiaries unrelated to its nuclear power plant ownership.

Exelon Corporation

Exelon Corporation was formed in 2000 by the merger of Unicom (Commonwealth Edison Company's parent) and PECO Energy Company. Commonwealth Edison's 10 operating nuclear plants have been transferred to Exelon Generation Company, LLC, ("EGC") which is a wholly owned subsidiary of Exelon Ventures Company, LLC, which, in turn, is a wholly-owned subsidiary of Exelon Corporation. PECo's Limerick and Peach Bottom nuclear plants also have been transferred to EGC, as has PECo's ownership interest in the two Salem Nuclear Plants.

PECo also owned 50 percent of the AmerGen Energy Company, LLC, ("AmerGen") which had acquired and operated three nuclear power plants in the U.S.: Three Mile Island Unit 1, Clinton, and Oyster Creek. PECo's interests in AmerGen have been transferred to EGC, LLC. Consequently, through EGC, LLC, Exelon Corporation owns and operates part or all of 16 nuclear plants and owns part of another three units.

The current organizational structure through which Exelon owns these nuclear assets is illustrated in Attachment No. 1 to this Report.

Entergy

Entergy Corporation was a pioneer in establishing separate corporate entities to own and operate nuclear power plants. Entergy today owns and operates ten nuclear units through an extensive network of wholly-owned subsidiaries.

Entergy currently owns five nuclear units in the South through five wholly-owned retail public utility companies and another wholly-owned subsidiary, System Energy Resources, Inc.[7]

Entergy also has purchased another five nuclear units in the Northeast including its just completed purchase of the Vermont Yankee nuclear plant. As shown in Attachment No. 2 to this Report, Entergy owns each of these units through a multi-tiered series of subsidiaries, many of which are limited liability companies. For example, the Indian

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[7] Entergy Arkansas, Inc., Entergy Gulf States, Inc., Entergy Louisiana, Inc., Entergy Mississippi, Inc., and Entergy New Orleans, Inc.

Point 2, Indian Point 3, and Fitzpatrick nuclear units are each owned by a separate LLC.[8] In the case of Indian Point 2, the immediate owner is Entergy Nuclear IP2, LLC. This company is, in turn, owned by Entergy Nuclear Investment Company III, Inc., which is a wholly-owned subsidiary of Entergy Nuclear Holding Company #3 that, in turn is a wholly-owned subsidiary of Entergy Nuclear Holding Company. Entergy Nuclear Holding Company, Inc., is a direct subsidiary of Entergy Corporation.[9]

The structure through which Entergy owns the Indian Point 3 and Fitzpatrick units is even more complex because each of the LLCs that owns these plants is, in turn, 50 percent owned by two other indirect Entergy subsidiaries, Entergy Nuclear New York Investment Company I and Entergy Nuclear New York Investment Company II. As shown in Attachment No. 2, these two Entergy Nuclear New York Investment Companies are themselves subsidiaries of Entergy Nuclear Holding Company #1 which, in turn, is a wholly-owned subsidiary of Entergy Corporation.

Another Entergy subsidiary, Entergy Nuclear Operations, Inc. ("ENO") operates Entergy's nuclear units in the Northeast.[10] Additional services are provided by other Entergy subsidiaries such as Entergy Services, Inc. (management, administrative and support services) and Entergy Nuclear Fuels Company (nuclear fuel planning, procurement and related services).

Entergy has provided the following explanation for this tiered holding company structure:

> Entergy Nuclear Holding Company, a first tier of Entergy Corporation, has been established with the intent that it will ultimately hold all the subsidiaries associated with Entergy's nuclear operations. This will consolidate all of Entergy's unregulated nuclear operations under a single holding company, while still supporting the operational and financing demands of the individual plants. *The use of holding companies below Entergy Nuclear Holding Company allows Entergy to segregate various types of financing, investment and business activities, and by doing so, enables Entergy to better manage and control risks associated with these activities.*[11] (Emphasis added)

Remarkably, Entergy has indicated that only two of all of the subsidiaries included in Attachment 2 -- ENO and Entergy Nuclear Generation Company, which owns and operates the Pilgrim Nuclear Station -- have any employees other than officers.[12] The

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8 Although the wholly-owned subsidiary that currently owns Entergy's Pilgrim Station is not an LLC, Entergy has said that it will seek to change the form of that subsidiary to an LLC in the near future.

9 Entergy has said that ultimately all of subsidiaries associated with Entergy's nuclear operations will be owned by Entergy Nuclear Holding Company. Rebuttal Testimony of Connie Wells, Entergy Nuclear Vermont Yankee, LLC, in Vermont Public Service Board Docket No. 6545, at page 9.

10 Entergy's nuclear units in the South are operated by yet another subsidiary, Entergy Operations, Inc.

11 Rebuttal Testimony of Entergy Nuclear Vermont Yankee witness Connie Wells in Vermont Public Service Board Docket No. 6545, dated February 25, 2002, at page 9.

12 Entergy response to Department of Public Service Information Request No. 2-10 in Vermont Public Service Board Docket No. 6545.

rest of the listed subsidiaries are merely paper organizations. In addition, the subsidiaries listed on Attachment 2 share many of the same individuals as officers.[13]

The NRC requires licensees of deregulated nuclear plants to provide certain financial guarantees that a unit would have sufficient funding to enable the licensee to continue to maintain the unit in a safe manner in case of an extended outage or a premature shutdown. Entergy's financial guarantees for its deregulated units in the Northeast are provided by two subsidiaries not listed in Attachment 2 -- Entergy International Holdings, LTD LLC and Entergy Global Investments, Inc. Both of these subsidiaries are themselves holding companies.[14]

As shown in Attachment No. 3, Entergy also has a very extensive network of other subsidiaries, in addition to those that own and operate its deregulated nuclear units in the Northeast.

Dominion

Dominion Resources, Inc. ("DRI") owns the two operating nuclear power plants at Millstone Point in Connecticut through a multi-tiered chain of subsidiaries. As shown on Attachment No. 4, DRI owns Dominion Energy Holdings, Inc. which, in turn, owns Dominion Energy Inc., LLC which owns Dominion Nuclear, Inc.. Dominion Nuclear, Inc. then owns Dominion Nuclear Marketing I, Inc, Dominion Marketing II, Inc, and Dominion Marketing III, LLC that together own Dominion Nuclear Connecticut, the direct owner of the Millstone nuclear station.[15]

Dominion also owns the four nuclear units at its North Anna and Surry stations in Virginia through the Dominion Generation Corporation which is a wholly-owned subsidiary of Dominion Energy Holdings, Inc. Dominion Generation Corporation also will own the fossil and hydro facilities that were formerly owned by Virginia Power Company.

Constellation

Constellation Energy Group, Inc. ("Constellation") purchased 100 percent of the Nine Mile Point Unit No. 1 nuclear plant and 82 percent of Nine Mile Point Unit No. 2 nuclear plant in 2001. Both of these units are located in upstate New York, near the City of Oswego. When Constellation sought NRC approval to transfer the units' licenses it also requested approval to complete a complex fourteen step corporate realignment. The nuclear-related results of this proposed realignment are shown on Attachment No. 5. The direct owner of the two Nine Mile Point nuclear plants is Nine Mile Point Nuclear Station, LLC, which is a wholly owned subsidiary of Constellation Nuclear Power Plants,

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13 Synapse has learned greater detail about Entergy's current holding company structure through its involvement on behalf o the Vermont Department of Public Service in Vermont Public Service Board Docket No. 6545.

14 Entergy response to Department of Public Service Information Request No. 1-42(c) in Vermont Public Service Board Docket No. 6545.

15 Dominion's August 17, 2001 letter to the NRC concerning the Millstone Nuclear Power Station Corporate Restructuring.

Inc, which, in turn, is a wholly owned subsidiary of Constellation Nuclear, LLC. Constellation's other two nuclear plants are owned by another subsidiary of Constellation Nuclear Power Plants, Inc, Calvert Cliffs Nuclear Power Plant, LLC. Constellation also has numerous other nuclear-related subsidiaries. Constellation Nuclear, LLC is, in turn, a subsidiary of Constellation Energy Group, Inc.

The parent corporation resulting from this corporate realignment will be BGE Corporation which will own Constellation Energy Group, Inc., as an immediate subsidiary.

Other Companies

The owners of fleets of nuclear power plants are not the only corporations that have established multi-tiered holding companies to own their nuclear plants. For example, as part of its proposed reorganization to recover from bankruptcy, Pacific Gas & Electric is seeking permission to transfer its two Diablo Canyon Nuclear Plants to a new LLC subsidiary, Diablo Canyon LLC. As shown on Attachment No. 6, this subsidiary would, in turn be a wholly owned subsidiary of Electric Generation, LLC, which in turn is a subsidiary of the Newco Energy Corporation, a wholly owned subsidiary of PG&E Corporation.[16]

Another example is Public Service Enterprise Group ("PSEG") which owns and operates the Salem and Hope Creek nuclear plants and is part owner of the Peach Bottom Nuclear generation station through a line of wholly-owned subsidiaries that includes PSEG Power, LLC, and its wholly-owned subsidiary, PSEG Nuclear.

Finding No. 3 - Limited Liability Companies are relatively new business structures that are used to shield the assets of a parent corporation from liability for financial risks.

The fundamental purpose and rationale for the creation of a "corporation" is to allow investors to pool their resources to engage in a business activity while limiting the financial consequences or "liability" of each individual investor. The most typical arrangement is for an investor to purchase stock or "shares" in the corporation. The money or other value paid for the shares is the limit of that investor's personal liability. The corporation's total liability is limited to the value of its investors' shares, plus any insurance policies that may be applicable.

Partnerships, an alternative form of business organization, are characterized by the inability of the partners to limit their individual liability. Each partner is wholly and personally responsible for all debts of the business. This onerous feature of partnerships has led to the development of many variations on the partnership model, particularly the limited partnership, as a way to shield some or all of the partners from unlimited liability.

Looking only at the liability issue, one might wonder why partnerships are ever chosen as a business structure. There are two primary reasons: streamlined management and lower

[16] PG&E's November 30, 2001 Application to the NRC for License Transfers and Conforming Administrative License Amendments.

taxes. A corporation is required to have articles of incorporation, a board of directors, and a management structure separate from the board of directors. Partnerships can be much more flexible with the same individual, or group of individuals, performing both day-to-day management and decision-making functions. Tax policy significantly favors partnerships by allowing all business profits to flow directly to the partners where they are taxed on their business income along with any other personal income. Corporations, because they are considered a separate entity, must pay corporate taxes before profits can flow to its investors, who then pay taxes on their corporate income on an individual basis. This is commonly called the "double taxation" feature of corporations.[17]

The dilemma facing entrepreneurs who want to start a business is whether the business structure should be designed to protect their existing personal assets by limiting their liability (a corporation) or whether the business structure should be designed to allow them to maximize their income from this single venture through lower taxes (a partnership). As discussed below, the nuclear industry seeks to achieve both liability protection <u>and</u> maximum income through the use of new limited liability corporate structures.

Limited Liability Subsidiaries

Limited liability companies (LLCs) are relatively new business structures that combine features of corporations and partnerships. An LLC has the same limited liability of corporations, but has the management flexibility of a partnership. Most significantly, pursuant to an IRS ruling in 1988, an LLC is considered a partnership for federal income tax purposes.

The first LLCs in the United States were formed in Wyoming in 1977 for foreign corporations that wanted to invest in very risky mineral exploration and development. Since 1977, LLC statutes have been enacted in all fifty states. They have proven to be a particularly attractive business structure for investments in high-risk ventures. LLCs can be formed by individuals, partnerships, or corporations. They can be managed by the LLC members (owners) or by an elected group of members, or by a single member. The management choice also acts to specify the members who can legally bind the LLC through contracts with outside entities.[18]

LLCs have become a very attractive business structure for corporations that acquire nuclear power plants. By creating a separate LLC for each nuclear plant, the profits from each plant's operations can flow back to the parent corporation without any intervening tax liability. The parent corporation's liability for each plant is limited to the investment the parent corporation made in initially setting up the LLC. Also, there can be more than one LLC between the parent corporation and the most risky component of the overall

[17] For general background on business structures, see any of a number of law school texts on business organizations. The information above is derived from "Organizing Limited Liability Companies: The Trend Continues", Richard M. Fijolek, Practicing Law Institute (1997); "A Limited Liability Company Checklist", Jerome P. Friedlander, II, Federal Lawyer (March/April 1995); and "The ABCs of LLCs, Steven Auderieth, Vermont Bar Journal and Law Digest (February, 1995).

[18] See above, Friedlander and Auderieth.

investment. For example, the technical support services for several nuclear plants can be consolidated into a separate LLC that contracts with all the individual plant LLCs. If one nuclear plant becomes unprofitable and goes into bankruptcy proceedings, in theory, only the single plant LLC assets are in jeopardy; the technical services LLC can continue to provide services to all the other single plant LLCs.

A particular concern regarding the use of LLCs is the situation where a parent corporation inserts several layers of LLCs between itself and the entity operating a high-risk business. Each of those intervening LLCs can act as a barrier to extending liability to the parent corporation that contains most of the assets. As noted in the case studies in Finding No. 2 of this Report, this approach appears to have been embraced by the parent corporations that recently have been purchasing nuclear plants. If a nuclear plant was unable to cover its liabilities, it might require several separate litigations, or a very large and complex single litigation, to pierce all the corporate veils back to the parent corporation with the bulk of the assets.

Finding No. 4 –There continue to be significant financial and other risks associated with nuclear power plant ownership and operations.

The restructuring of electricity markets has meant increased risks for owners of any deregulated electric generation facilities, whether their plants are fossil-fired or nuclear. Revenues which used to be based on traditional "cost of service" concepts and stable rates are now based instead on the actual sales from a power plant at market prices that are sometimes volatile.

At the same time, there are significant nuclear-related risks that could have a material adverse effect on nuclear power plant owners. For example, a recent Prospectus issued by Exelon Corporation for the sale of $700 million of notes by Exelon Generation Company, LLC specifically identified the following risks associated with owning and operating nuclear power plants:

> We may incur substantial cost and liabilities due to our ownership and operation of nuclear facilities. The ownership and operation of nuclear facilities involve certain risks. These risks include: mechanical or structural problems; inadequacy or lapses in maintenance protocols; the impairment of reactor operation and safety systems due to human error; the costs of storage, handling and disposal of nuclear materials; limitations on the amounts and types of insurance coverage commercially available; and uncertainties with respect to the technological and financial aspects of decommissioning nuclear facilities at the end of their useful lives. The following are among the more significant of these risks:
>
> <u>Operational risk.</u> Operations at any nuclear generation plant could degrade to the point where we have to shut down the plant. If this were to happen, the process of identifying and correcting the causes of the operational downgrade to return the plant to operation could require significant time and expense, resulting in both lost revenue and increased fuel and purchased power expense to meet our supply commitments. For plants operated by us but not wholly

> owned by us, we could also incur liability to the co-owners. We may choose to close a plant rather than incur substantial costs to restart the plant.
>
> Regulatory risk. The NRC may modify, suspend or revoke licenses and impose civil penalties for failure to comply with the Atomic Energy Act, the regulations under it or the terms of the licenses of nuclear facilities. Changes in regulations by the NRC that require a substantial increase in capital expenditures or that result in increased operating or decommissioning costs could adversely affect our results of operations or financial condition.
>
> Nuclear accident risk. Although the safety record of nuclear reactors generally has been very good, accidents and other unforeseen problems have occurred both in the United States and elsewhere. The consequences of an accident can be severe and include loss of life and property damage. Any resulting liability from a nuclear accident could exceed our resources, including insurance coverages.

These same risks apply to other nuclear plants including those owned and operated by multi-tiered holding companies and LLCs.

The industry's expressed desire to build new nuclear plants also can be expected to increase the financial pressures on licensees as they may have to further reduce O&M expenditures at existing plants in order to fund the construction of new ones.

Finding No. 5 - The NRC has expressed concern that deregulation can adversely affect the safety of operating nuclear power plants by increasing the pressure on licensees to reduce costs.

Although it has been said that an efficient and economical plant is often a safe plant,[19] the NRC has expressed concern that the transition to economic deregulation can adversely affect nuclear power plant safety and may not provide the same degree of assurance that adequate funds would be provided for safe operation and decommissioning.[20]

The NRC has further explained the impact that increased competition can have on nuclear power plant economics and safety:

> As described in SECY-97-253, traditional "cost-of-service" regulation, under which virtually all NRC power reactor licensees have operated, has typically been effective in providing necessary funds for licensees to operate and decommission their nuclear plants safely. With the advent of greater competition within the electric utility industry, pressures to reduce costs and improve efficiency have increased and will almost certainly intensify as deregulation proceeds. Moreover, with deregulation of the generation sector of the industry, traditional cost-of-service regulation is likely to essentially disappear for most generators. Thus, the concept of electric utility, as

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[19] *NRC Staff Requirements Memorandum, SECY-98-153*, dated June 29, 1998, at page 3.

[20] *NRC Final Policy Statement on the Restructuring and Economic Deregulation of the Electric Utility Industry* (62 *Fed. Reg.* 44071; August 19, 1997)

currently defined in 10 CFR 50.2 may in the future no longer be meaningful for a large number of, if not all, power reactor licensees. Electricity rates set by competition in a free market may not provide the same degree of assurance of adequate funds for safe operation and decommissioning as traditional cost-of-service ratemaking. In SECY-97-253, the staff cited the example of the "Independent Safety Assessment of Maine Yankee Atomic Power Company" (NRC Staff Report: Ellis W. Merschoff, Team Lead; October 1996), which concluded, "Economic pressure to be a low-cost energy producer has limited available resources to address corrective actions and some plant improvement upgrades.

When the NRC issued its Final Policy Statement on the Restructuring and Economic Deregulation of the Electric Utility Industry (62 *Fed. Reg.* 44071; August 19, 1997), specific safety concerns with respect to rate deregulation and restructuring were identified. For example, the final policy statement discussed such safety concerns as reductions in expenditures for manpower and training and other reductions in operations and maintenance (O&M) and capital additions budgets. The issues of on-line maintenance and increased fuel burnup were also addressed.

In addition, with respect to specific plants such as Maine Yankee, Millstone, and others, the inspection process has identified several manifestations of inappropriate responses to competitive pressures. These include: increased need for corrective actions; maintenance operator work-arounds; temporary modification and procedure revision backlogs; decreased performance in operator licensing and requalification programs; increased frequency of significant operational and occupational safety events; decreased plant and system reliability; increased volume and acrimony of allegations; and increased frequency of regulatory violations and resulting penalties.

As deregulation proceeds, cost pressures may increase these types of reductions in safety margins at plants. Moreover, because the impact of budgetary reductions can cut across all plant safety-related programs, other impacts in addition to those previously identified may occur as a result of deregulation. For example a merchant plant with no assets other than the nuclear plant itself could be unable to make necessary safety expenditures after an extended outage if it did not have an adequate financial cushion to pay costs incurred during the outage. In such a situation, it is not clear that a transition from indefinite shutdown to permanent shutdown and decommissioning would be sufficiently smooth to prevent funding shortages from causing safety problems during the shutdown transition period. That is, given the requirements in 10 CFR 50.82 with respect to: (1) the limitation on the use of the trust fund for legitimate decommissioning activities; and (2) the timing of significant decommissioning trust fund withdrawals, a licensee could run out of funds for operational safety expenses before it was able to draw on its decommissioning trust fund. This, in turn, could force the NRC to make the decision for the licensee to permanently cease operations and initiate decommissioning pursuant to 10 CFR 50.82.[21]

The nuclear industry itself has acknowledged the safety and economic risks associated with economic deregulation. For example, a former President of the industry's American Nuclear Society told the Society's Winter 2001 meeting that "Safety is the highest priority because of the impact on cost that would result from an NRC-forced shutdown" and that there is now "actually a higher focus on safety than before."[22] However, he also noted the challenges that come from deregulation and restructuring:

> With restructuring comes challenges for plant operators and regulators, Quinn continued. These challenges for operators include management focus on economics, not safety; pressure on workers to keep plants operating (because of volatility of electricity prices); pressure to reduce preventative maintenance; deferral of equipment replacements; and less investment for safety backfits. For the regulator, these include increased workload (because of mergers, license transfers, etc.); pressure to avoid requiring shutdowns of plants; and increased political pressure to reduce the regulatory burden. Challenged also is the nuclear technology infrastructure. According to Quinn, there is less cooperation among competing nuclear utilities, and less safety research and technical support for the plants.[23]

Finding No. 6 - The NRC has expressed concern that the use of holding company structures can reduce the assets that would be available for the safe operation and decommissioning of a nuclear power plant. However, the NRC does not adequately protect against the risk that an LLC subsidiary will transfer all of its operating profits to its parent company or engage in risky loans to or questionable deals with affiliates.

The NRC Staff has expressed concern that the use of holding company structures can lead to a diminution of the assets necessary for the safe operation and decommissioning of a licensee's nuclear power plant.[24] In fact, as early as March 1993 the NRC Staff expressed concern that:

> Current and potential organizational structures of many power reactor licensees and their corporate affiliates are complex and evolving. The staff believes that the public health and safety implications of such structures warrant further examination. A licensee subsidiary without assets other than

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21 *NRC Staff Requirements Memorandum, SECY-98-153*, dated June 29, 1998, at pages 2 and 3.

22 *ANS Winter Meetings: Nuclear Power - Attracting Notice, A Brighter Outlook,* Nuclear News, August 2001, starting at page 34.

23 Ibid.

24 *Safety Evaluation by the NRC's Office of Nuclear Reactor Regulation "Related to Proposed Corporate Restructuring of Commonwealth Edison Company,"* October 5, 2000, at page 3.

> the licensed reactor could renege on its decommissioning obligations if forced to shut down prematurely. Given that corporate law generally limits the liability of stockholders, the NRC may not have recourse to the assets of a parent company if its subsidiary defaults absent legally enforceable commitments by owners. Case law with respect to bankruptcy proceedings is also ambiguous. Although bankruptcy courts have generally directed bankruptcy trustees to make justifiable, legally required expenditures to protect public health and safety, it is not clear that these expenditures will always have a high priority relative to other claims. The staff believes that it should evaluate possible ways to increase assurance of decommissioning funds availability. An increased degree of confidence may be appropriate to assure that the problems that the Office of Nuclear Material Safety and Safeguards has had with some of its licensees abandoning materials sites prior to cleanup will not be experienced for power reactor licensees.[25]

The NRC Staff consequently requested that the NRC Commissioners approve publication of an advance notice of proposed rulemaking to explore alternatives to mitigate the potential impact on safety of power reactor licensee ownership arrangements and to consider whether increase assurance of funding availability for decommissioning activities was needed.

> *A licensee subsidiary without assets other than the licensed reactor could renege on its decommissioning obligations if forced to shut down prematurely.*
>
> NRC Staff, March 1993

Unfortunately, the NRC Commissioners disapproved this request and, instead, asked for additional information on the staff proposal. In response to a Commission question on how many reactor licensees could try to set up a corporate veil to avoid decommissioning costs, the NRC Staff noted:

> Potentially, any investor-owned utility could establish a holding company to which it could transfer the bulk of its assets over time. If forced to shut down prematurely, a licensee with assets limited essentially to the shut down reactor could declare bankruptcy and renege on any unfunded decommissioning obligation. If a bankrupt licensee had insufficient assets, a bankruptcy court might be powerless to order that assets of a parent company be used to fund decommissioning, even if the court wished to do so.[26]

In the years since 1994, the NRC has not developed or adopted any policy limiting the transfer of operating profits from the subsidiary that directly owns a nuclear plant. Nor

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25 *Issuance of An Advance Notice of Proposed Rulemaking on the Potential Impact on Safety of Power Reactor Licensee Ownership Arrangements, SECY-93-075,* March 24, 1993, at page 1.

26 *Response to Staff Requirements Memorandum of April 28, 1993, Which Disapproved Issuance of An Advance Notice of Proposed Rulemaking on the Potential Impact on Safety of Power Reactor Licensee Ownership Arrangements, SECY-94-280,* at pages 4 and 5

does the NRC have any policy limiting the types or magnitudes of the loans that such an operating subsidiary can make to affiliated companies.

At most, the NRC merely conditions license transfer approvals to new holding company structures upon a requirement that the licensee not transfer to its proposed parent or any other affiliated company significant assets for the production, transmission or distribution of electric energy without first notifying the NRC. The NRC has defined "significant assets" to be facilities having a "depreciated book value exceeding 10% of the company's consolidated net utility plant."[27]

The NRC also does not have a specific policy statement or procedure on how limited liability companies or other types of licensees use financial assurance funds in the forms of lines of credit for plant operation.[28] Nor does the NRC have any specific policy statement or procedure that controls how it would consider approval of requests of limited liability companies to reduce, replace, or withdraw available lines of credit that are subject to NRC conditions. Instead, the NRC has said that it will review such requests on a case-by-case basis.[29]

The NRC has explained its policy for addressing situations where a licensee has drawn upon the lines of credit provided by a parent or affiliated companies. In such situations, the NRC would:

> evaluate the reasons behind [the licensee's] drawing on the lines of credit. The staff cannot provide a detailed discussion of potential agency actions until it learns the specific reasons for the usage of such funds. Generally, if drawings on the lines of credit were made to cover short-term cash flow deficiencies that did not appear to have any significant safety ramifications, the NRC would not likely need to take any specific action. If drawing on the lines of credit were to indicate serious longer-term financial problems that appeared to potentially adversely impact protection of public health and safety, the NRC would monitor the effects of any degradation on protection of public health and safety and act appropriately.[30]

The NRC's failure to have any policy limiting the transfer of operating profits from the subsidiary that directly owns a nuclear plant or the types or magnitudes of the loans that such an operating subsidiary can make to affiliated companies is all the more significant because the new holding companies also may have not set policies governing these matters. For example, Entergy has said that there are no written procedures governing the distribution of operating profits from the subsidiaries that are the direct owners of its

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27 For example, see the October 5, 2000 Safety Evaluation by the NRC Office of Nuclear Reactor Regulation of the proposed corporate restructuring of PECO Energy Company, at page 3.

28 Enclosure 1 to the NRC's December 13, 2001 letter to Christine Salembier, Commissioner, Vermont Department of Public Service, on the subject of "Vermont Yankee Nuclear Power Station – Lines of Credit Associated with Vermont Yankee License Transfer."

29 Ibid.

30 Ibid.

nuclear units.[31] These subsidiaries either make distributions to their immediate parent companies or make loans to affiliated companies depending on the specific cash requirements of the parent companies or the affiliates.

Vermont Department of Public Service witness Andrea Crane has explained to the Vermont Public Service Board why it should be concerned about the ability of a parent corporation to drain the funds available to a nuclear power plant-owning subsidiary:

> ...in addition to being concerned about the availability of capital for ENVY's[32] operations, there is also a concern that Entergy Corp. may threaten the long-term financial viability of ENVY by using ENVY's earnings to fund other Entergy Corp. operations, leaving insufficient funds in ENVY for nuclear operations. Therefore, in addition to raising concerns about the availability of sufficient operating and capital funds, I am also concerned about the need to retain capital in ENVY. The Board should avoid a repeat of the situation that transpired in PG and E ... whereby funds were transferred from a successful operating entity to the holding company, leaving the operating company in dire financial straits.[33]

Ms. Crane also expressed concern about the absence of formal Entergy corporate policies governing the transfer of profits and inter-affiliate transactions:

> The lack of direct control over its internally generated funds, and the vagueness of the corporate policy, does not provide an appropriate level of financial assurance for the ownership and operation of a nuclear power plant. It leaves open the possibility that Entergy Corp could require 100% of operating earnings as dividends from its subsidiaries, including ENVY, if it needed funds to meet other priorities or emergencies, leaving the owners of the nuclear plants without sufficient capital to pursue their own immediate priorities.[34]

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31 Entergy Response to Department of Public Service Information Request No. 2-36 in Vermont Public Service Board Docket No. 6545.

32 ENVY is Entergy Nuclear Vermont Yankee LLC, which is the Entergy Corporation subsidiary that will own the Vermont Yankee nuclear plant if the purchase is approved by the Vermont Public Service Board.

33 Direct Testimony of Andrea Crane on behalf of the Vermont Department of Public Service, Vermont Public Service Board Docket No. 6545, at page 9.

34 Direct Testimony of Andrea Crane on behalf of the Vermont Department of Public Service, Vermont Public Service Board Docket No. 6545, at page 28.

Finding No. 7 - The NRC's reviews of the financial qualifications of new nuclear power plant owners are inconsistent and may be too limited to ensure that subsidiaries will have adequate funds to safely operate and decommission their nuclear plants and pay retrospective Price-Anderson Act premiums.

Before it allows a nuclear power plant operating license to be transferred, the NRC conducts reviews of the financial qualifications of the prospective owner. The NRC's regulations specify the types of information that a prospective licensee must provide and the nature of the review that must be conducted by the NRC staff.

However, the applicable NRC regulation, 10 CFR 50.33(f), is inconsistent in that on the one hand it says that "the applicant shall submit information that demonstrates the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operation costs for the **period of the license.**" (emphasis added) But the regulation then merely requires applicants to submit estimates for total annual operating costs for only the first 5 years of operation of the facility. Although the NRC can ask for information for subsequent years, this regulation can mean that the NRC will only review five years of operating cost data when the new owner may be seeking transfer of a license which will continue in effect for another 25 years or longer.

In reviewing the financial qualifications of a prospective licensee, the NRC requires that the new owner either meet a supply and demand test or show that it has an investment grade rating or equivalent from at least two bond-rating organizations. The supply and demand test examines whether the prospective licensee will earn sufficient revenues (either from the sale of electric power from the nuclear plant or from other sources) to cover expected operational expenses at the plant.[35] This analysis is based on the applicant's uncertain and speculative estimates of total operating revenues and costs for the first full five years following the expected completion of the license transfer.[36] At the same time, it is very unlikely that the new corporate subsidiaries that actually will own the transferred plant will have issued any securities that had received investment grade or equivalent ratings from any bond-rating organizations.

If a prospective licensee is unable to meet either the supply and demand test and or the bond rating criteria test, the NRC will consider its ability to fund a six-month outage. Although assuring the funding for a six-month outage is not required where a prospective licensee meets either of the NRC's two primary tests, in those cases where a prospective licensee voluntarily guarantees the funds to pay for a six-month outage, the Commission will accept that commitment and impose a licensee condition prohibiting the applicant from voiding or diminishing those guarantees.

The U.S. General Accounting Office ("GAO") has evaluated the NRC's review of the financial qualifications of prospective licensees to safely operate and decommission

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[35] NUREG-1577, Revision 1, at Section III.1.b.

[36] It also appears that the NRC does not consider the need to pay retrospective Price-Anderson Act premiums when it considers a prospective licensee's financial qualifications to safely operate and decommission a nuclear power plant.

nuclear power plants. The GAO concluded that for the most part, the NRC's reviews of new owners financial qualifications have enhanced the level of assurance that they will safely own and operate their plants in a deregulated environment and not need to shut them down prematurely.[37]

However, the GAO also found that the NRC did not always adequately verify the new owners' financial qualifications.[38] In particular, the GAO concluded that when the NRC reviewed the financial qualifications of Exelon to safely own and operate the largest fleet of nuclear plants in the U.S., it did not require the same additional guarantees from the parent or affiliated companies that the new owner would have sufficient revenues to cover the plants' operating costs as it had required from other proposed license transfers.[39] The NRC also did not validate the information submitted by the new owner to demonstrate that the company was financially qualified.[40] In fact, the GAO concluded that the NRC had eventually transferred the licensees to Exelon Generation Company on the basis of projected financial information that both the affected companies and the NRC knew to be inaccurate.[41]

The NRC's review of financial qualifications continues after a license is transferred. Each licensee is required to submit an annual financial report, pursuant to 10 CFR 50.71(b) and a decommissioning funding status report is required every two years.[42] The NRC Staff also monitors the general financial status of nuclear plant licensees by screening the trade and financial press reports, and other sources of information.[43]

However, it is unclear whether the NRC has the staff resources or the expertise to conduct adequate reviews of licensee's financial qualifications. For example, the NRC's Executive Director for Operations informed the Commissioners in April 1997 that the expertise of the NRC Staff in matters of finance and economic analysis were "limited."[44] At the same time, the size of the NRC Staff has been reduced by approximately ten percent since 1997.[45]

The NRC has expressed confidence in its Staff's ability to identify financial distress and has quoted approvingly a Staff member who said "severe financial distress from any of the licensees is something that's not going to be hidden from view very long."[46] However, the suddenness of ENRON's collapse and the apparent absence of public warnings of that

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37 *Nuclear Regulation: NRC's Assurances of Decommissioning Funding During Utility Restructuring Could be Improved*, GAO-02-48, December 2001, at page 6.

38 Ibid., at page 4.

39 Ibid., at page 21.

40 Ibid., at pages 21 and 31-32.

41 Ibid., at page 33.

42 10 CFR50.75(f)(1).

43 NUREG-1577, Rev 1, Section III.1.d., at page 5.

44 NRC SECY-97-071, April 2, 1997.

45 NUREG-1350, Vol. 13, Figure 4.

46 *In the Matter of Power Authority of the State of New York and Energy Nuclear Fitzpatrick,* 53 N.R.C. 488, June 21, 2001.

company's severe financial distress prior to that collapse suggest that the NRC may not have any warning about a licensee's impending financial problems.

Finally, the NRC recently has indicated its intention to reduce the regulatory burden on licensees by eliminating the requirement that licensees include financial qualifications information in license renewal applications.[47] This would mean that there would be no assessment of the financial qualifications of a licensee to safely operate a nuclear power plant for up to an additional twenty years beyond the expiration of its existing NRC-issued license.

In conclusion, there are a number of reasons to have serious concerns about the quality of the NRC's review of the financial qualifications of licensees and prospective licensees.

Finding No. 8 - The financial guarantees that the NRC requires from prospective nuclear power plant owners may not be adequate to assure that plants are operated and decommissioned safely and that plant owners will be able to pay deferred Price-Anderson Act insurance premiums in the event of a nuclear accident.

The NRC has generally accepted guarantees from prospective nuclear power plant licensees in the range of $55 to $75 million to pay for a six-month outage. However, in a number of cases the licensee has not offered and the NRC has not required the licensee to make any such guarantee.[48] For example, there appears to be guarantees in place for only three of the nuclear units owned by Exelon Generation Company, LLC. These are the three units that were originally 50 percent owned by PECO Energy Company and were transferred to Exelon Generation Company, LLC as part of the merger between Unicom and PECO Energy. The guarantees that were in place when the plants were owned by PECO Energy and British Energy were transferred along with the plants. However, it does not appear that there is any guarantee in place for the other 16 nuclear plants that are currently owned by Exelon Generation, Company, LLC.

There is no evidence that these limited $55 to $75 million guarantees will provide sufficient funds to enable power plant owners to safely shutdown their nuclear plants in case of a serious event or significant problem and to maintain the plant in a safe shutdown condition until the problem is addressed or the licensee is able to gain access to the plant's decommissioning trust fund. For example, a substantial number of nuclear power plants have been shutdown since January 1996 for outages that lasted far longer than six months:

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[47] FedNet Government News, June 5, 2002.

[48] As we will discuss in Finding No. 9, Constellation has guaranteed that its nuclear power plant-owning subsidiaries, Nine Mile Point LLC and Calvert Cliffs Nuclear Plant LLC will receive whatever cash is needed to protect the public health and safety.

Table No. 2
Nuclear Power Plant Outages
Since June 1995
That Lasted Nine Months or Longer

Plant	Period Shutdown	Outage Duration
Beaver Valley 2	December 1997 - September 1998	9 months
Clinton	September 1996 - May 1999	32 months
Cook Unit 1	September 1997 - December 2000	39 months
Cook Unit 2	September 1997 - June 2000	33 months
Indian Point 2	February 2000 - December 2000	10 months
Kewaunee	September 1996 - June 1997	9 months
LaSalle Unit 1	September 1996 - August 1998	23 months
LaSalle Unit 2	September 1996 - April 1999	31 months
Millstone Unit 2	February 1996 - May 1999	39 months
Millstone Unit 3	March 1996 - June 1998	27 months
Point Beach Unit 1	February 1997 - December 1997	10 months
Point Beach Unit 2	October 1996 - August 1997	10 months
Salem Unit 1	May 1995 - April 1998	35 months
Salem Unit 2	June 1995 - August 1997	26 months

Indeed, as Table No. 1 (in Finding No. 1 above) and Table No. 2 reveal, it is not unusual for more than one unit at a single site to be shutdown for an extended outage at the same time. These simultaneous extended outages could significantly increase the financial pressures on the units' owner in a deregulated environment when its cash flow depends on the actual sales from the plant rather than on regulated rates for an entire utility.

Moreover, it is not unreasonable to expect that a nuclear unit might be shutdown for more than six months before the ultimate parent corporation makes the decision to permanently retire the unit. After all, the full extent of the plant's problems and the expense and time it would take to repair and restart the unit might not be apparent until the plant had been shut down for a substantial period of time.

This could mean that all of the funds guaranteed by an affiliate or the parent corporation could be exhausted before the licensee would be able to gain access to the unit's decommissioning fund. For example, Millstone Unit 1 was shutdown for 31 months before Northeast Utilities decided in July 1998 to permanently retire the plant. Commonwealth Edison Company's Zion Units 1 and 2 were shutdown for eleven and sixteen months, respectively, before the Company decided in January 1998 to permanently retire both plants. The Maine Yankee plant was shut down for eight months before its Board of Directors decided in August 1997 to permanently retire it.

But even if an outage were shorter than six months, the maintenance and/or capital expenditures required to repair a plant and restore it to service may be significantly higher than the company had projected in its application to the NRC. The limited funds pledged by a parent corporation or an affiliate could be inadequate under such circumstances.

Finding No. 9 – The NRC has proposed to significantly reduce its review of a non-electric utility licensee's financial qualifications when it evaluates an application to renew a nuclear plant's operating license.

The NRC has proposed to eliminate the requirement that non-electric utility power reactor licensees submit financial qualifications information in their license renewal applications.[49] At the same time, the NRC also has proposed to require the submission of such information when utilities reorganize and operate as "non utility" generators.

The NRC's proposal to require financial reviews when a utility recognizes with a new financial structure is important. However, the decision to reduce disclosure obligations on nuclear power plant owners when they seek renewal of operating licenses for up to 20 years creates the potential for added risk of non-performance in critical areas.

A formal and rigorous review at the time of license renewal for aging nuclear reactors is a particularly appropriate time to evaluate the financial requirements. It is at this point that a business plan can be evaluated over the proposed lifetime of a licensee's facility. The financial resources needed to address the safe and secure operations, make capital improvements to a complex 30 year old machine, meet added license conditions required after the events of September 11, 2001, and to meet decommissioning and public liability obligations under the Price Anderson Act, must be juxtaposed against the economic conditions in the electricity markets and the availability of capital and insurance.[50]

The NRC's justification for not requiring a financial qualifications review at the time of relicensing is that it can monitor licensees when changes take place in licensee's financial qualifications. These day-to-day or limited annual reviews are not substitutes for a

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49 June 4, 2002 Federal Register, at Vol. 67, No. 104, pp. 38427.

50 The wisdom of looking into the future was underscored in the case of USEC, Inc, which has an NRC Certificate under 10 CFR Part 76 to operate uranium enrichment plants. The NRC conducted a financial review of the USEC, Inc. Certificate when it was issued in 1998 using the threshold of a current investment grade credit rating. The NRC determined USEC was reliable and economic based on its BBB+ investment grade debt rating. However, the NRC did not look beyond the 5 year term of the certificate to evaluate USEC's financial qualifications or the company's ability to operate with an unsustainable business model. If it had, it could have readily foreseen that USEC's financial condition would deteriorate over time due to a number of factors including the declining value of its sales contract, lower market prices, increasing unit costs of output and lack of competitive technology to enrich uranium for nuclear power reactors. These factors led to multiple credit downgrades and subsequent NRC doubts about whether USEC's economic resources were sufficient to be recertified for another 5 years.

formal, rigorous and disciplined review examining all a licensee's financial ability to fulfill its obligations for safely and securely operating an aging reactor in a competitive marketplace.

Historically, the ratemaking process for a utility corporation had provided reasonable assurance that a license applicant would have funds necessary to operate a reactor. In these circumstances, a licensee could be assumed of obtaining all of the reasonable funds it needed to continue operating its aging power plant. However, non-utility generators now lack the same assured funding, and as utilities diversify into telecommunications, trading operations and high-risk financial activities, the risk that there will be insufficient capital grows. To provide a green light for 20 years of operation without a rigorous review of a licensee's financial resources and business plans invites unwelcome surprises.

Finding No. 10 - The NRC does not require that parent corporations guarantee that funds will be provided to safely operate and decommission the nuclear power plants owned by their subsidiary companies.

The NRC does not require that a parent corporation guarantee the funds that may be needed to operate and decommission safely the nuclear power plants owned by subsidiaries. Instead, the NRC Staff has included conditions requiring a parent guarantee in the orders approving license transfers as additional assurance of financial qualifications only when such a guarantee has been offered by the applicant.[51].

For example, in its reviews of the financial qualifications of Entergy Corporation to own the Pilgrim, Indian Point 2, Indian Point 3, Fitzpatrick, and Vermont Yankee nuclear plants, the NRC has accepted guarantees that would be provided through lines of credit from affiliated financial subsidiaries which may not have sufficient liquid capital when it is needed by a plant-owning affiliate. One of these credit lines is to be used for working capital, if needed. The other is not intended to be used in the normal course of business but instead would be used in the event of problems at the plant. Entergy has indicated that this line of credit would be used to pay the costs between the unplanned shutdown of a plant and the availability of funds from the plant's decommissioning trust fund.[52]

Vermont Department of Public Service witness Andrea Crane has explained the problems that can arise from the fact that neither of the Entergy subsidiaries that provide these lines of credit have any physical assets:

> The result is that these two companies are only as strong as 1) their receivables from, and investment in, associated companies, and 2) Entergy Corp's commitment to provide them with additional funds, if required. Entergy Corp, therefore, has full discretion as to whether or not to provide sufficient capital to EIHL and EGI so that these two financing vehicles can

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[51] *In the Matter of GPU Nuclear, Inc and AmerGen Energy Company, LLC*, 51 N.R.C. 193, at Footnote No. 8.

[52] Prefiled Direct Testimony of Michael R. Kansler, Entergy Nuclear Vermont Yankee, Vermont Public Service Board Docket No. 6545, at page 10.

> meet their commitments to ENVY. If Entergy Corp should choose to walk away from EIHL and EGI, there appears to be no recourse for ENVY.[53]

For this reason, Ms. Crane recommended that the parent Entergy Corporation be required to guarantee that the pledged funds actually would be available if needed:

> Entergy Corporation should be obligated to stand behind the total financial exposure occasioned by the ownership and operation of this nuclear power plant. It is not reasonable to allow Entergy Corporation to shield itself from financial responsibility with complex financial arrangements. It certainly should not be allowed to offer guarantees from subsidiaries that do not have sufficient assets to meet their obligations on a stand-alone basis, because the parent could walk away from those subsidiaries if its own interests so dictated. If Entergy Corporation intends to stand behind the guarantees of its subsidiaries, it should have no problem in making the guarantee directly.[54]

Even though the NRC had accepted the $70 million guarantee provided by the two lines of credit from Entergy Corporation subsidiaries, in response to the concerns raised by the Ms. Crane and the Vermont Public Service Board, the parent Entergy Corporation has provided an additional financial guarantee of up to $60 million.[55] As Entergy has explained:

> The intent of that guarantee is to make sure that, in the event of a premature shutdown of the Vermont Yankee Nuclear Power Station, there will be money available to bridge the gap between shutdown and the point at which ENVY is able to access the decommissioning trust fund. Thus, if either line of credit has been drawn upon, Entergy will guarantee to make up any deficiency up to a total of $60 million.[56]

Entergy also acknowledged that the parent corporation has not provided a similar guarantee in support of **any** of the other nuclear plants its subsidiaries have acquired.[57] It further noted that state and federal regulators, including the NRC, had found the smaller guarantees by affiliated companies, not the parent corporation, to be sufficient.[58]

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53 Direct Testimony of Andrea Crane on behalf of the Vermont Department of Public Service, Vermont Public Service Board Docket No. 6545, at page 18.

54 Direct Testimony of Andrea Crane on behalf of the Vermont Department of Public Service, Vermont Public Service Board Docket No. 6545, at page 22.

55 Ms. Crane subsequently testified that the revised commitments by the parent Entergy Corporation adequately addressed the concerns in her Direct Testimony. Supplemental Testimony of Andrea Crane in Support of the Memorandum of Understanding in Docket No. 6545, at page 2 of 9.

56 Rebuttal Testimony of Connie Wells, Entergy Nuclear Vermont Yankee, LLC, in Vermont Public Service Board Docket No. 6545, at page 3, lines 8-13.

57 Ibid., at page 5, lines 1-5.

58 Ibid.

Dominion has voluntarily committed $150 million from the parent corporation, DRI, to assure that Dominion Nuclear Connecticut (the new owner of the Millstone Nuclear Station) will have sufficient funds available for meeting its operating expenses for the recently acquired Millstone Units 2 and 3.[59] Dominion has explained that the subsidiary, Dominion Nuclear Connecticut, has the right to obtain such needed funds from DRI as it determines "are necessary to protect the public health and safety, meet NRC requirements, meeting ongoing operational expenses or to maintain Units 2 and 3 safely."[60] However, it does not appear that Dominion has made the same commitment to the four nuclear plants at the Surry and North Anna sites owned by the Dominion Generation Corporation.

Constellation has guaranteed that each of its nuclear power plant-owning subsidiaries, i.e., Nine Mile Point Nuclear Station LLC and Calvert Cliffs Nuclear Plant LLC, would be provided whatever cash is needed to protect the public health and safety.[61]

But it does not appear that the parent Exelon Corporation has guaranteed any funds to its power plant owning and operating subsidiary Exelon Generation Company, LLC.

Finding No. 11 – Taxpayers may be at risk if nuclear plant owning subsidiaries are unable to continue making safety-related or decommissioning expenditures or pay retrospective Price-Anderson Act premiums.

In attempting to assure the Vermont Public Service Board that the former owners of the Vermont Yankee nuclear plant and their ratepayers are unlikely to be required to pay any shortfalls in decommissioning funds, Entergy has noted that the NRC has on several occasions said that the burden of paying any such shortfalls would fall on taxpayers:

> NRC regulations do not specifically address the potential liability of other parties in the event that the licensed owner is unable to provide the funds required for decommissioning. In the past, the NRC indicated that any failure of the licensed owner to meet its decommissioning funding obligations would result in a burden on taxpayers -- presumably in the form of a publicly funded cleanup. See, e.g., SECY-94-280 (Nov. 18, 1984), at 4. ("Such action would either increase the potential risk to public health and safety of the decommissioning process or would shift the burden of decommissioning funding from ratepayers to taxpayers.") (emphasis added); 61 Fed. Reg. 15427, 15428 (Apr. 8, 1996)("The liability of the licensee to provide funding for decommissioning may adversely affect protection of the public health and safety. Also, a lack of decommissioning funds is a financial risk to taxpayers

59 *Dominion August 31, 2000 Application for the transfer of the licenses for Millstone Units 1, 2 and 3*, at page 10.

60 Ibid.

61 *Calvert Cliffs Nuclear Power Plant Request for a Transfer in Control*, December 20, 2000, at page 9 and *Nine Mile Point Unit Nos. 1 & 2 NRC License Transfer Application*, February 1, 2001, at page 23.

(i.e., if the licensee cannot pay for decommissioning, taxpayers would ultimately pay the bill. (emphasis added)."[62]

In fact, there are a number of possible circumstances in which taxpayers could be asked to bear much, if not all, of the cost of a major power plant accident. First, there is no assurance that the primary tier of insurance would be available to a licensee in the event of an act of terrorism against a nuclear power plant. American Nuclear Insurers has testified that it would only have resources available to provide the primary insurance coverage to cover a single act of terrorism.[63] Thereafter, all licensees would be left without any primary insurance coverage. At that point, licensees might seek recourse in the courts for a finding that domestic terrorism is an "act of war." Acts of war are excluded from coverage under the Price-Anderson Act.[64]

At the same time, the liabilities associated with a nuclear accident are borne by every nuclear power plant owner in the U.S. as a result of the pooling of liabilities for accidents with claims in excess of $200 million. The maximum cost per reactor is $88.085 million (subject to inflation adjustments) in secondary liability. As shown on Table No. 3 below, the liability for nuclear owners with multiple plants, such as Exelon (19 units) and Entergy (10 units), could approach or exceed $1 billion.

Table No. 3
Potential Price Anderson Act Nuclear Insurance Liabilities

Parent Corporation	Maximum Potential Annual Liability	Maximum Potential Total Liability
Exelon Corporation	$163.52 million	$1,440.35 million
Entergy Corporation[65]	$99 million	$872.04 million
Duke Energy	$52.50 million	$462.45 million
Dominion Resources, Inc.	$57.03 million	$502.32 million
Southern Company	$39.16 million	$344.94 million

62 Legal Memorandum on the "Decommissioning Liability Associated with a Power Reactor License," Goodwin Procter LLP, February 24, 2002, submitted by Entergy Corporation to the Vermont Public Service Board as Exhibit ENVY-Wells-3 to the Prefiled Rebuttal Testimony of Connie Wells in Docket No. 6545.

63 John Quattocchi, Senior Vice-President, American Nuclear Insurers, February 15, 2002 Response to Question from Senator Reid, Hearing before the Senate Committee on Environment and Public Works, January 23, 2002.

64 The NRC's "opinion" is that claims arising out of an act of terrorism at a nuclear power plant would not be excluded under the Price Anderson Act. February 13, 2002 NRC Answer to Question No. 3 from Senator Reid, Hearing before the Senate Committee on Environment and Public Works, January 23, 2002. However, the NRC recognizes that a "question of this nature and magnitude" would likely need to be resolved by a court in the first instance.

65 Potential Liability figures reflect Entergy ownership of the Vermont Yankee Nuclear Station.

TVA	$60 million	$528.51 million
Progress Energy	$44.72 million	$393.87 million
FPL Group[66]	$47.33 million	$416.91 million
Constellation Energy Group, Inc.	$38.20 million	$336.49 million
FirstEnergy	$40 million	$352.34 million

However, under the Atomic Energy Act, a licensee's secondary liability can be deferred if it would constitute an undue hardship on the licensee.[67] In such a situation, the secondary liability that would have been borne by the license would become a taxpayer funded liability. It is not unreasonable to expect that power plant owners, especially those that are thinly capitalized, will try to avail themselves of this deferral should a major accident occur.

Moreover, this Report has focused on nuclear-related issues. Nuclear power plants also contain large amounts of asbestos and large volumes of toxic chemicals. Taxpayers also could be forced to bear the costs of cleaning up for these and any other non-nuclear-related pollutants if a single asset power plant-owning subsidiary was able to successfully declare bankruptcy and a court was unwilling to hold the parent corporation liable.

Finding No. 12 - The NRC has no statutory authority to require a licensee in bankruptcy to continue making safety-related or decommissioning expenditures or to pay retrospective Price-Anderson Act premiums.

NRC regulations require any nuclear power plant licensee to immediately report any filing of a voluntary or involuntary petition for bankruptcy.[68] However, the NRC has no additional financial requirements for situations where a licensee files for bankruptcy or otherwise encounters financial difficulties. Nor does the NRC have any statutory authority to require a licensee which is in bankruptcy to continue to make safety-related or decommissioning payments or to pay retrospective Price-Anderson Act premiums. The NRC must intervene in the proceedings before the bankruptcy court and petition the court to require such payments.

The NRC has acknowledged that the license transfer requirements contained in 10 CFR 50.80 do not specifically or expressly refer to a prospective licensee's ability to meet financial protection payments that may be required under the Price-Anderson Act.[69] However, the NRC has said that 10 CFR 140.21 requires reactor licensees that are covered under the Price-Anderson system to provide annual guarantees of payments of

[66] Potential Liability figures reflect FPL Group ownership of the Seabrook Nuclear Station.

[67] Atomic Energy Act Section 170(b)(2)(A) and (2)(B).

[68] 10 CFR 50.54 (cc).

[69] NRC February 13, 2002, response to Post-Hearing Question 6 from Senator Reid.

retrospective premiums and that the NRC evaluates an applicants guarantees of payment of retrospective premiums when it considers a license transfer request. [70]

The NRC has further said that it annually reviews the Price-Anderson Act guarantees for all of its power reactor licensees, including those that are LLCs.[71] All of the licensees have, to date, used the cash flow method of guarantee allowed under 10 CFR 140.21; that is, a licensee may demonstrate that it has sufficient cash flow over 3 months to meet an annual $10 million retrospective premium payment for each reactor that it owns.[72] As long as the licensee chooses that method and is able to pass the financial test for cash flow each year, no additional guarantee is required. However, if a licensee were not able to pass the cash flow test, it would have to provide some other allowable guarantee such as surety bonds, letters of credit, revolving credit/term load arrangements, maintenance of escrow deposits of government securities, or such other type of guarantee as might be approved by the NRC.[73] But there is no requirement that the parent corporation provide such a guarantee, only the subsidiary, and there is no requirement that resources be available to pay the maximum of $88.085 million per reactor.

The NRC has stated that under 10 CFR 140, a licensee is required to pay the retrospective premium, notwithstanding its financial status.[74] The NRC also has said that its has had positive experiences with bankruptcy courts that have overseen the Chapter 11 reorganizations of Public Service Company of New Hampshire (Seabrook nuclear plant), Cajun Electric Cooperative (River Bend), El Paso Electric Company (Palo Verde), and Vermont Electric Generation & Transmission Cooperative (Millstone 3).[75] According to the NRC, in each of these cases, the bankruptcy courts allowed these bankrupt licensees to pay all safety-related operational and decommissioning expenses (including, the NRC believes, Price-Anderson primary layer and on-site property insurance premium payments). The NRC also has noted that during its bankruptcy PG&E has continued to meet all safety-related expenses for its nuclear plants.

However, the NRC has acknowledged that it could potentially face a conflict with other claims in a bankruptcy proceeding "if there were an accident sufficient to trigger a retrospective premium assessment. The NRC would presumably require a licensee to pay the assessment, but the bankruptcy court could order the licensee not to pay it."[76]

In addition, the NRC's earlier experience with the bankruptcies all involved entities that owned a number of different assets. The bankruptcy of a single-asset LLC, which owns only a single nuclear power plant, would present very different circumstances and

[70] NRC February 13, 2002, response to Post-Hearing Question 6 from Senator Reid.

[71] The NRC also requires that each licensee submit an annual financial report, 10 CFR 50.71(b) and a decommissioning fund status report every two years (and annually during the last five years of operation). 10 CFR 50.71(f)(1).

[72] A retrospective premium is insurance that is paid after an accident.

[73] NRC February 13, 2002, response to Post-Hearing Question 8 from Senator Reid.

[74] NRC February 13, 2002, response to Post-Hearing Question 9 from Senator Reid.

[75] NRC February 13, 2002, response to Post-Hearing Question 2 from Senator Inhofe.

[76] NRC February 13, 2002, response to Post-Hearing Question 9 from Senator Reid.

challenges. At the same time, as we will discuss later in this Report, given the multi-tiered holding companies (including LLCs) through which parent corporations now own many nuclear power plants, the NRC might have trouble "piercing the corporate veil" to require a parent of a bankrupt LLC subsidiary to make the required retrospective premium payments.

It is clear that there are no specific statutory or regulatory safeguards in place to ensure that retrospective premiums under the Price-Anderson Act will be available from bankrupt nuclear plant-owning subsidiaries or from their parent corporations. The NRC has sought legislation from Congress to ensure that decommissioning costs receive explicit priority in bankruptcy proceedings. But, so far, that legislation has not been enacted.[77] The NRC has further stated its willingness to support legislation to prioritize safety-related claims in bankruptcy proceedings and to avoid any potential conflict between NRC requirements to pay into the retrospective Price-Anderson Act premium pool and other claims in bankruptcy.[78]

Finding No. 13 – Case law suggests that it would be very difficult to hold a parent corporation responsible for the liabilities incurred by nuclear power plant owning LLC subsidiaries in a multi-tiered holding company.

As mentioned earlier in this Report, the multiple layers of subsidiaries, including LLCs, that have been created by parent corporations in the nuclear industry are a cause of serious concern. Even if a court concludes that the liability of the subsidiary that actually operates the nuclear plant should be extended to business structures above it (for example, if under capitalization and profit distributions have left the subsidiary unable to cover the costs of unanticipated repairs or security improvements and the subsidiary decides to cease operations), the ability of the court to find a senior business entity with sufficient capital could be complicated by multiple layers of subsidiaries and LLCs. There may be issues of jurisdiction, applicable state or federal statutes, the role of the NRC, and other myriad issues of law and fact that would need to be resolved. Given that the presumption in every state and federal statute is for the <u>limitation</u> of corporate liability, the burden is always on the party trying to extend that liability to show that the law, facts, and public policy all support violating the statutory presumption.[79] Courts, in

[77] The Energy Policy Act of 2002 (HR 4), as approved by the U.S. Senate, amends the U.S. Bankruptcy Code to prevent creditors in a bankruptcy proceeding from attaching an NRC licensee's decommissioning funds until the decommissioning has been completed. The Senate enacted provision also seeks to prevent creditors from using Price-Anderson insurance and those deferred premiums held in reserve to satisfy creditors. However, neither version of the Energy Policy Act of 2002, that enacted by the House or the Senate, would require a parent corporation or other guarantor to commit resources in the event that there are not adequate resources within a bankrupt LLC to satisfy claims after a nuclear accident. Post accident liabilities could shift to taxpayers in this case.

[78] NRC February 13, 2002, response to Post-Hearing Question 9 from Senator Reid.

[79] "Piercing the Corporate Veil: An Empirical Study", Robert B. Thompson, 76 Cornell Law Review 1036 (1991), Section II, and "Limited Liability and the Corporation", Frank H. Easterbrook and Daniel R. Fishel, 52 U. Chi. L. Rev. 89 (1985), Section IV.

general, are reluctant to pierce the corporate veil and extend liability; when multiple corporations are involved, that reluctance only increases.

Despite the limitations on corporate liability embodied in statutes, there are numerous instances where courts have been willing to ignore those limitations under a wide range of factual circumstances. The case law varies a great deal from one state to another, but all of them involve some rationale for "piercing the corporate veil" and holding the owners of the corporation personally liable. For the purposes of this Report, it is important to note that in the nuclear power industry, the owners of a nuclear power plant-owning LLC subsidiary are most likely to be another LLC or a parent corporation. The objective of the effort to pierce the corporate veil in this situation would be to make the parent corporation responsible for the liability of the LLC subsidiary.

There is an enormous volume of litigation over the issue of extending liability through to the owners of a corporation. The case law is varied and complex and a thorough and complete review is beyond the scope of this project. What follows is a summary of the common themes that have been used by a variety of courts for extending liability.[80]

Starting from a presumption that a corporation's liability is limited, facts must be presented to justify extending liability. Some of the fact situations that have been persuasive to courts are the following:

- Corporate form is used as a front for illegal or fraudulent activity. In these cases, courts express no reluctance in holding individuals liable for the debts of the corporation since there is no public policy that seeks to support such activities under any business structure.
- Corporate form is used as a sham or a mere shell to avoid liability. In these cases, the individuals or parent corporation are aware from the start that the corporation is unlikely to ever repay its debts or liabilities and seek to acquire as much income as possible before creditors foreclose.
- Individual owners subvert the corporation for their personal gain. In these cases, the personal enrichment may or may not be based on illegal or unethical actions. If the facts establish that owners personally benefited from corporate activities (beyond the normal sharing of corporate profits), then courts are generally willing to make them personally liable. These cases often involve members of the Board of Directors or managers. Corporate owners who do not personally benefit but are aware of the enrichment of other owners can be held personally liable based on their breach of their fiduciary responsibilities to the corporation.
- Under-capitalization of the corporation. In these cases, there is a determination that at the time of incorporation, or due to subsequent management actions, there is insufficient capital available for the business activities of the corporation. Although similar to the problem of a sham corporation, the decision by the court

[80] Id., Thompson at 1063-1072; Easterbrook at 109-113.

involves a more objective analysis of appropriate levels of capitalization for similar entities engaged in similar activities.

- Improper distributions of income. These cases involve decisions by the corporate management, usually the Board of Directors, to distribute corporate income to shareholders in a financially irresponsible manner that leaves the corporation unable to meet its obligations. These are very fact-specific litigations that involve a great deal of hindsight analysis. However, if the facts show a clear pattern of irresponsibility, as opposed to poor business decisions, courts will extend liability to specific individuals or the corporation in general.

- Interference in management. These cases involve situations where owners, often large stockholders in closely held corporations, become so involved in corporate management that they look more like a managing partner than just an investor. Courts will extend liability to these "investors" on the theory that they do not deserve the normal corporate protection.

- Environmental, regulatory, or public policy. These factors are often included with one or more of the above fact patterns to support extending liability. It is unusual for a court to invoke "public policy" by itself as a justification for piercing the corporate veil.

An empirical study of court decisions where piercing the veil issues were litigated indicates that courts are very reluctant to impute liability to the shareholders of public corporations. Closely-held corporations (non-public and usually with few investors) and related corporate entities (subsidiaries, affiliates, etc.) are the forms to which courts have applied extended liability.[81]

There is very little case law involving LLCs that specifically addresses piercing the corporate veil due to the relatively short time period (fifteen years) during which LLC structures have been developed. Consequently, there is great uncertainty as to the effect that having one or more LLCs in the ownership chain within a holding company will have on the willingness of a court to pierce the corporate veil in order to hold a parent company responsible for the liabilities of its indirect nuclear power plant owning-subsidiary.

Finding No. 14 - The NRC has expressed serious doubts as to its ability to hold a parent corporation responsible for the liabilities incurred by a subsidiary.

There are two NRC cases that involved attempts to pierce the corporate veil of the operator of a nuclear power plant. In 1995, the NRC attempted to negate a transfer of assets from a licensee which, as part of a complicated corporate restructuring, had become a subsidiary to a newly created holding company because the transfer had occurred without the prior written consent of the NRC, as required by section 184 of the Atomic Energy Act. The NRC held that it could pierce the veil of corporations that

[81] Id., Thompson at 1070.

violate section 184. However, before a final adjudication, this case ended in a settlement.[82]

In 1997, the NRC tried to force a parent company to provide additional funds to the decommissioning fund for a subsidiary plant. However, prior to a final adjudication, the NRC approved a settlement that resolved the decommissioning fund issue without any specific finding as to the parent company's liability.[83] In accepting the settlement, the NRC expressed concern that there was a "substantial possibility of defeat if the case proceeds to trial [on a theory of] piercing the corporate veil."

Both cases were cited in a legal memorandum provided by the current owners of the Vermont Yankee Nuclear Power Corporation, which concluded that attempts to pierce the corporate veil of nuclear power plant subsidiaries were unlikely to succeed and have seldom been attempted.[84] Despite the numerous specific instances where courts have extended liability to parent corporations, there is great uncertainty as to whether or not courts would apply such extended liability to multi-layered nuclear power companies.

Finding No. 15 – Shielding parent corporations from nuclear power plant operating and decommissioning risks is unfair and economically inefficient.

To the extent that the organizational structures discussed above serve to successfully shield the parent company from risks, they are inequitable and undermine efficient decision-making.

As a matter of fairness, individuals and companies should take responsibility for cleaning up after themselves. If an unanticipated problem in operation causes a nuclear plant to experience an extended or permanent outage prior to the end of its operating license or if the decommissioning of a plant turns out to cost more than expected, then the parent company may decide to provide additional resources to the subsidiary in order to carry out the subsidiaries responsibilities. On the other hand, the parent company may not. If there are clean up costs which the subsidiary is unwilling to bear, then these may fall upon taxpayers. Considerations of fairness would have the company that profited (or expected to profit) from plant operation bear the costs of cleaning up the facility.

This is also a matter of economic efficiency. If a company is protected from significant risks associated with its decisions, then there is what economists call an "externality." A reasonable definition of externality is provided in a popular economics textbook as follows:

> An externality or spillover effect occurs when production or consumption inflicts involuntary costs or benefits on others; that is, costs or benefits are

[82] *Safety Light Corp.*, 41 N.R.C. at 457-458 (1995).

[83] *Sequoyah Fuels Corp. and General Atomics*, CLI-97-13, 46N.R.C. 195 (1997).

[84] Vermont Yankee Memorandum of Law Regarding Ratepayer Risk of Liability for Vermont Yankee Decommissioning Costs, Vermont Public Service Board Docket No. 6545, dated February 25, 2002, at pages 17 and 18.

> imposed on others yet are not paid for by those who impose them or receive them. More precisely, an externality is an effect of one economic agent's behavior on another's well-being, where that effect is not reflected in dollar or market transactions.[85]

Where there are such externalities, private decision-making will be inefficient. A company will tend to undervalue (or value at zero) the costs associated with its action that are borne by others. In the case of a nuclear power plant, the protection from liability may, for example, cause the operator to make decisions that undervalue the potential for long-term radioactive waste storage costs. Or, faced with operating decisions that involve tradeoffs between cost and safety, the owner may undervalue safety and make choices that strike the wrong balance. In these situations, because some of the risks are "external," the market outcome may be an inappropriate decision from a societal perspective – or an inefficient allocation of resources. Government policy efforts should aim to internalize externalities, in order to promote appropriate private decision-making and efficient resource allocation.

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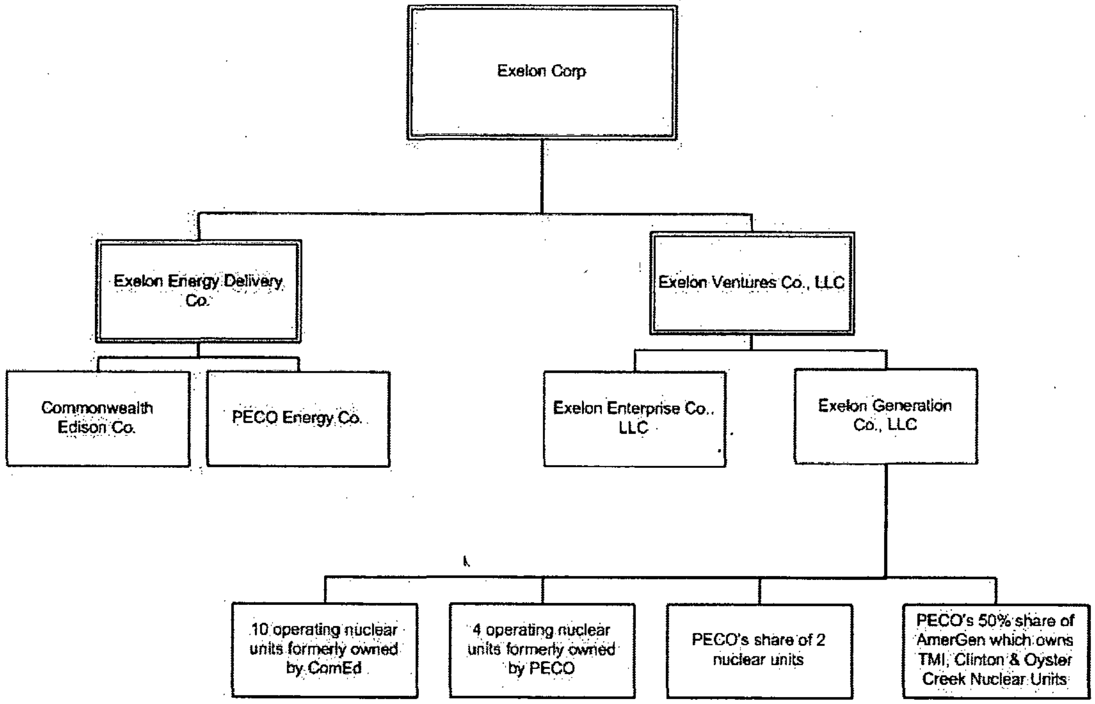
85 Samuelson, Paul A. and William D. Nordhaus. 1989. Economics, 13th Edition. McGraw-Hill, at page 770.

ATTACHMENT NO. 1

Exelon Corporation

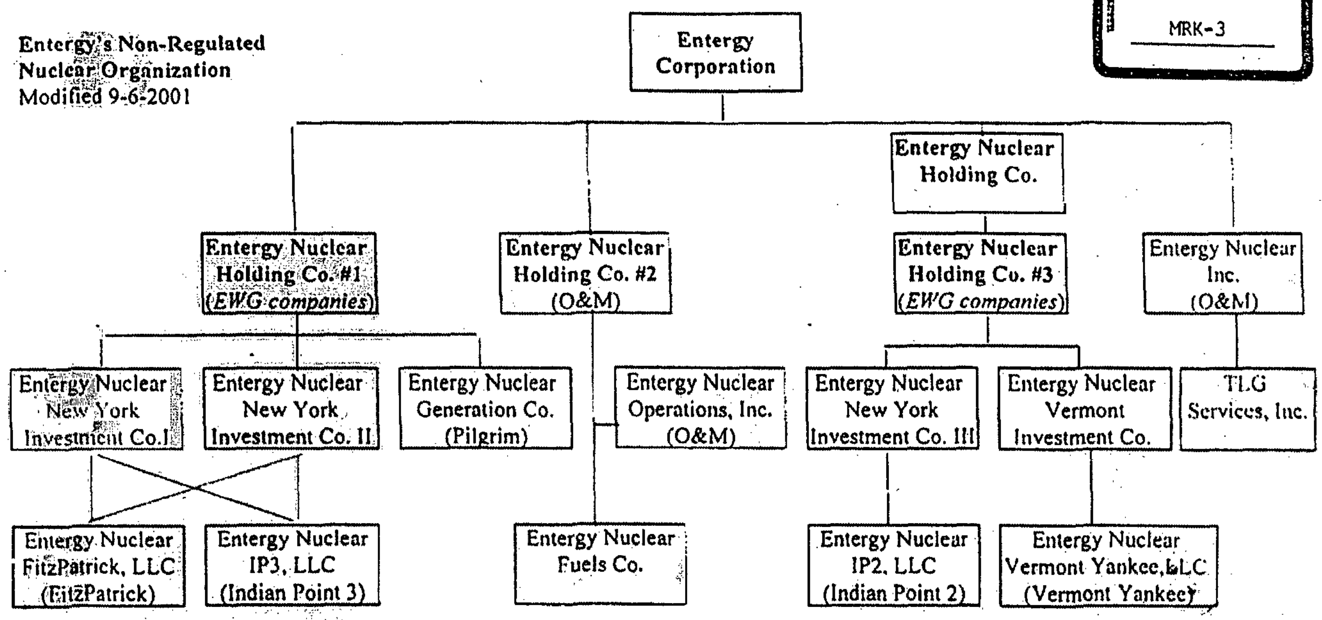


ATTACHMENT NO. 2

Entergy Corporation – Non-regulated Nuclear Organization

EXH. .T
MRK-3

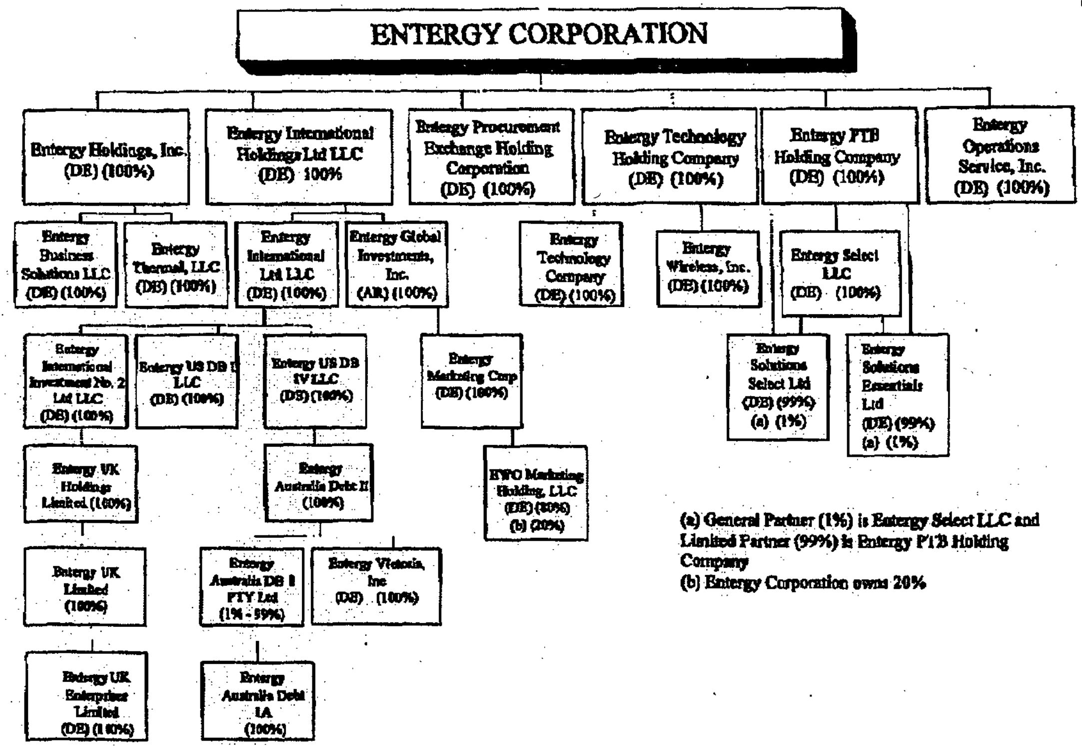
Entergy's Non-Regulated Nuclear Organization
Modified 9-6-2001



Exempt Wholesale Generator (EWG)
Operations and Maintenance (O&M)

ATTACHMENT NO. 3

Entergy Corporation

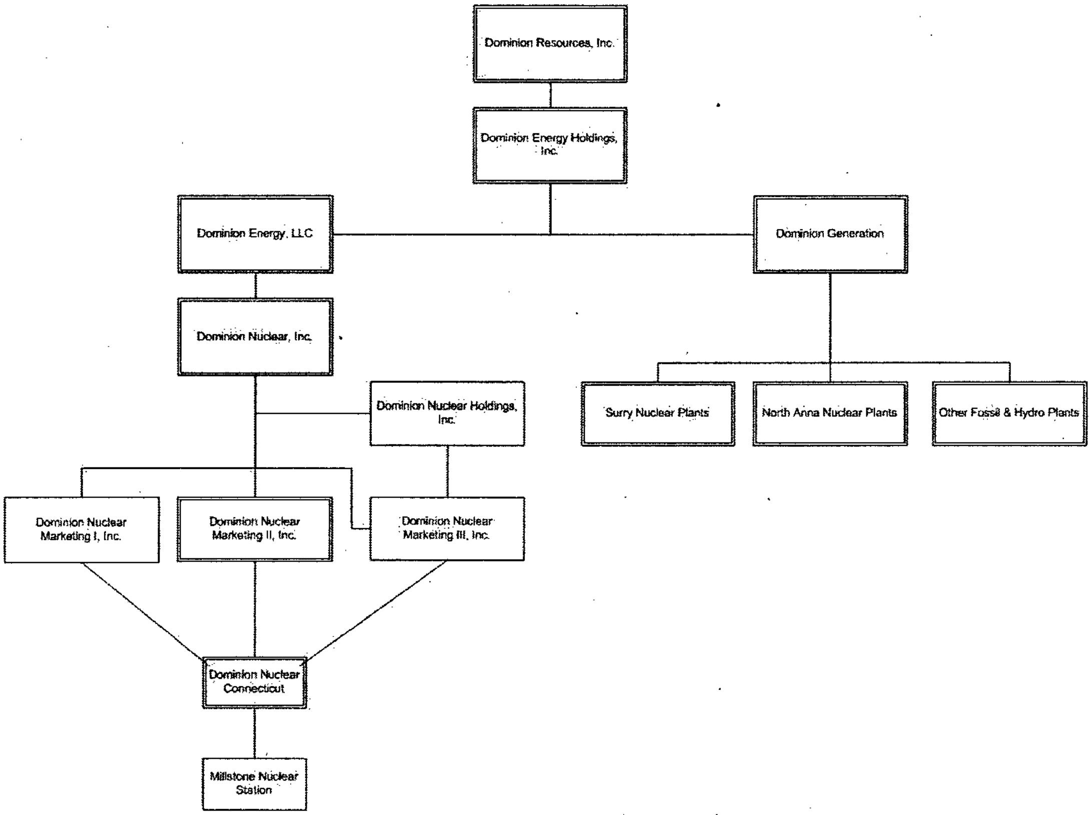


(a) General Partner (1%) is Entergy Select LLC and Limited Partner (99%) is Entergy PTB Holding Company
(b) Entergy Corporation owns 20%

Caldwhamlorg.chart\Chart #2 11-5-01.vsd

ATTACHMENT NO. 4

Dominion Resources, Inc.



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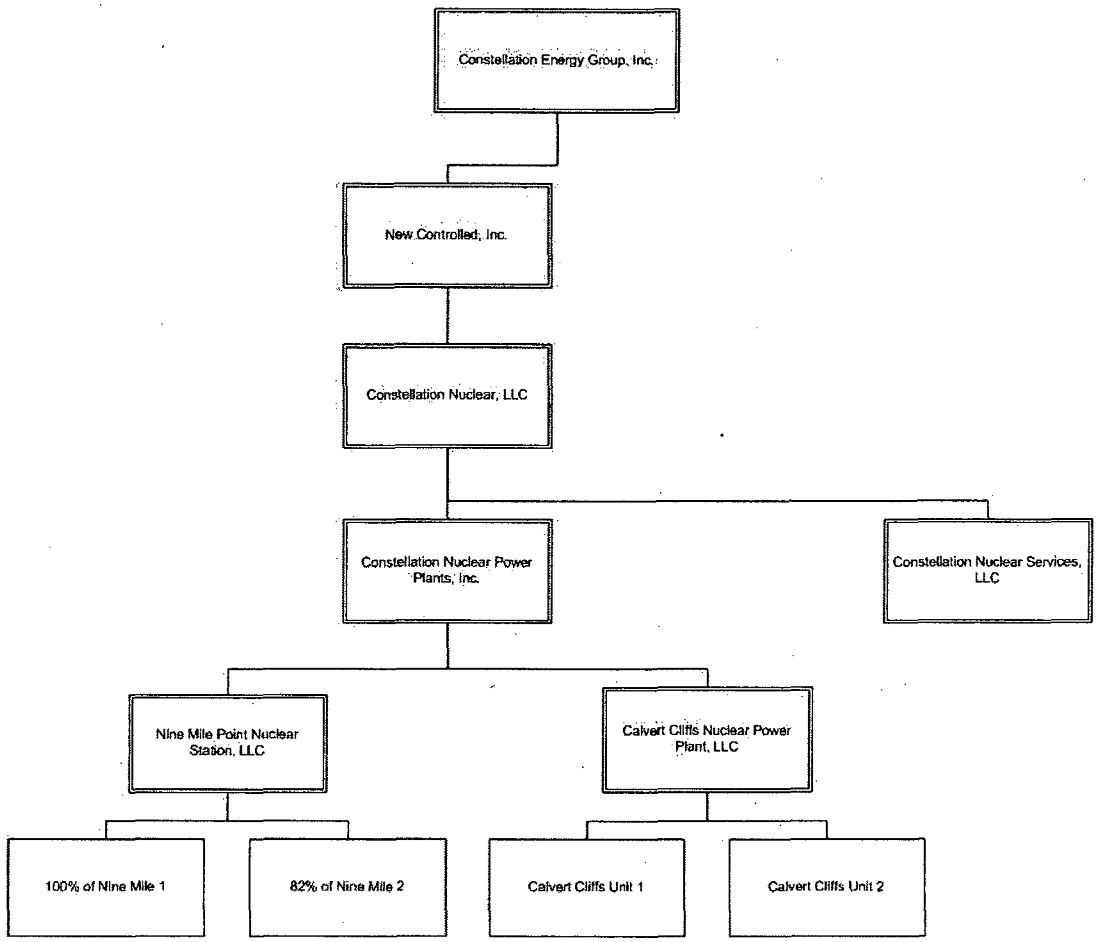
Constellation Energy Group



ATTACHMENT NO. 6

PG&E Corporation

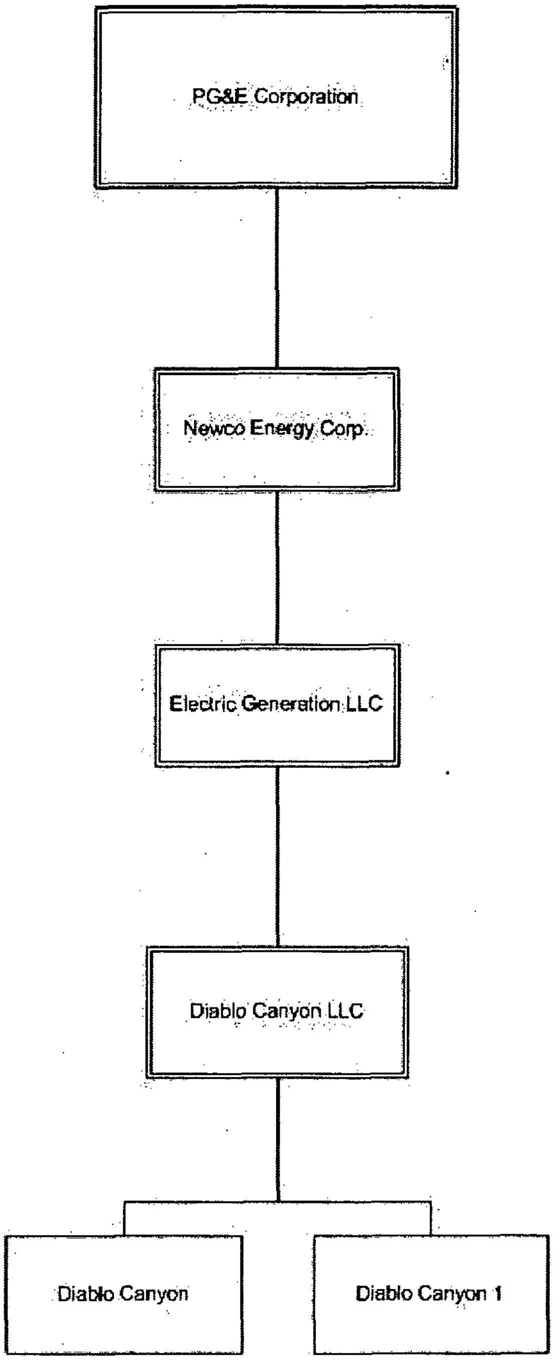


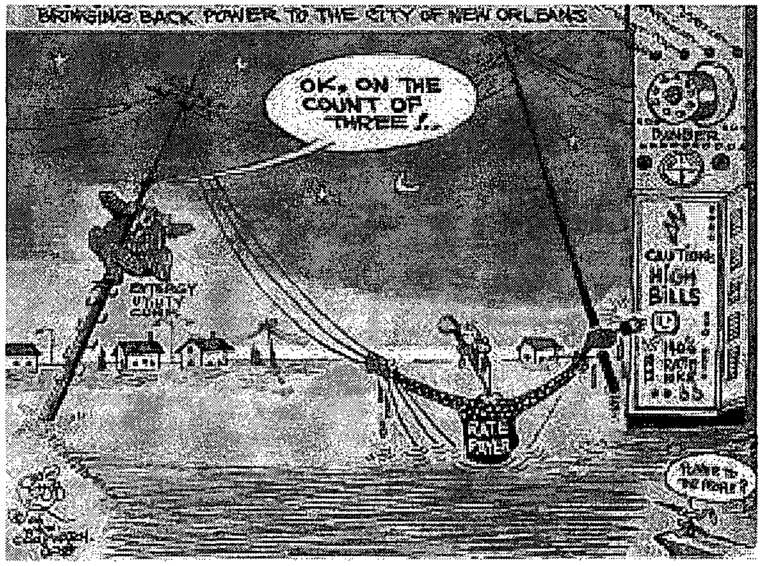
EXHIBIT W

CORPWATCH
Holding Corporations Accountable
www.corpwatch.org

Entergy Holds New Orleans for Ransom

by Rita J. King, Special to Corp Watch
May 10th, 2006

Some New Orleanians desperately want, and fear, their utility bills.



cartoon by Khalil Bendib

But Tom Morgan, a DJ at New Orleans' local radio station WWOZ, hasn't seen one in months. While many residents have been hobbled by the cost of having a certified electrician reestablish electric and gas connections to their homes, some — like Morgan — have been unable to find out how much they owe and fear a gigantic bill they won't be able to pay, meaning – to add insult to injury – they face having their power cut off.

"I haven't gotten a bill since October," Morgan told CorpWatch. "I've called Entergy, but it still hasn't come. You shouldn't have to chase after people to pay your bills. A lot of people are confused and afraid."

Cutting and Running?

Entergy Corp. was the last Fortune 500 Company still based in New Orleans before Hurricane Katrina struck. Unless the federal government grants its wholly owned subsidiary, Entergy New Orleans, the $718 million it seeks to maintain and rebuild its gas and electricity infrastructure damaged in the storm, the utility might not be able to continue doing business in the city it currently supplies with gas and electric. It isn't that Entergy can't afford to rebuild; it's that it would rather keep its profits and let the federal government pick up the tab.

Entergy Corp. racked up $10 billion in revenues last year and has $29 billion in collective assets. On paper, there is no question Entergy could comfortably cover its losses and rebuild the infrastructure of its utility business in New Orleans. On May 2, Entergy announced that its first-quarter profit rose nearly 13 percent, as higher energy prices offset disrupted sales following last year's hurricanes. Entergy CEO J. Wayne Leonard received a $1.1 million bonus at the end of 2005, according to SEC records, which coincidentally works out to one dollar per Entergy customer in the Gulf Coast left without power in the weeks following the hurricane.

But the company's executives feel that if anyone should pay the cost of its getting back into business, it should be ratepayers and taxpayers, and not its own shareholders. And indeed, the government may have little choice but to give in to what critics characterize as blackmail or extortion – or leave a major American metropolis powerless.

Gordon Howald, utilities analyst with New York-based Natexis Bleichroeder, said he hasn't warned clients to sell off their Entergy stock. When Entergy first announced the possibility of bankruptcy, he says he thought the claim was a bluff.

"Entergy is a pretty successful company making a lot of money and here you have all these people who have lost their homes," he said. "Months later, they're still bickering. As time goes on, it becomes more difficult to get recovery, and ratepayers will have to pick up the rest."

Using the Federal Government as Insurance

The story of Entergy in New Orleans is a cautionary tale, critics say, of privatizing utilities as critical as gas and electricity in major population centers. Although Entergy is regulated by the New Orleans City Council, and its customer base is the local citizenry, as a publicly held company it ultimately answers first to its shareholders who want to maximize their profits. To that end Entergy has made broad use of limited liability laws to structure the company and its subsidiaries in a way that insulates shareholders from liabilities such as storms. The result is a system whereby the company's own customers and taxpayers nationwide foot the bill when something goes wrong.

Entergy has estimated its losses post-Katrina at just over $1 billion, including lost customers (many residents who fled may not come back), and miles of gas pipeline corroded by the saltwater that poured over the levees. The insurance the company carried covered $250 million in damages. The rest will likely come from inflated rates for those customers who remain in New Orleans, and from federal funds.

The Federal Energy Regulatory Commission (FERC), which has minimal oversight of Entergy, does not require companies like Entergy to carry insurance to cover losses from catastrophic events such as a hurricane, even though conventional wisdom has long considered a Katrina-sized storm and flood inevitable. The cost of the insurance it does carry may be passed on to ratepayers, according to FERC spokesman Bryan Lee.

Citing a "precedent" set by bailouts of ConEdison and the airline industry after 9/11, Entergy New Orleans is seeking a Community Development Block Grant (CDBG) to make up the difference. Without CDBG aid, Entergy New Orleans estimates that the average ratepayer's share of the losses comes to $8,943, which would be exacted in the form of a rate increase of at least 140 percent.

No Cost of Doing Business in Hurricane Alley

In the days after the storm, all public comments focused solely on restoration, as if the effort to rebuild New Orleans could be accomplished with nothing more than elbow grease and well-wishing. President George W. Bush cut an iconic pose in New Orleans' Jackson Square on September 15, 2005, with his sleeves rolled up for a prime-time television appearance as he stood in front of a statue of Andrew Jackson and vowed "one of the largest reconstruction efforts the world has ever seen."

Meltdown? Not Our Problem

An Entergy subsidiary, Entergy Nuclear Northeast, operates the Indian Point nuclear power plant in Westchester County, New York. In late March, a huge crowd assembled to meet with Nuclear Regulatory Commission (NRC) and Entergy representatives, who repeatedly claimed that the numerous radioactive leaks recently detected seeping from the plant and toward the Hudson River were nothing to worry about. But the crowd was concerned, largely because they knew Entergy was positioned to protect its assets by filing for bankruptcy in the event of a nuclear disaster. Where, they wondered, was the motivation to maintain safety standards, if not the specter of lost profits?

"Corporations like Entergy have the luxury of walking away from these disasters unscathed, while citizens who lose their homes and jobs will be saddled with the additional burden of having to pay for Entergy's mess," said Lisa Rainwater, Indian Point campaign director for watchdog environmental group Riverkeeper. "The new consumer bankruptcy law is yet another disturbing example of how Washington is more interested in protecting the financial interests of corporate America than the financial security of hard-working Americans. It's appalling."

What has been happening in New York with Entergy Nuclear may provide clues about what's ahead for New Orleans. Activists have been sounding the alarm there for years about the potential for a private company to cut and run with its profits, leaving devastation – and a massive bill – in its wake.

Indeed, the nuclear industry is no stranger to federal assistance. In July 2003, the Bush-Cheney Energy Bill was passed by Congress, and Republican Senator John McCain referred to it as the "no lobbyist left behind" bill. It was considered by many to be a bonanza for the

"Throughout the area hit by the hurricane," Bush pledged, "we will do what it takes, we will stay as long as it takes, to help citizens rebuild their communities and their lives." Two weeks later, on October 4 in the Rose Garden at the White House, Bush had a new philosophy: cut non-security spending to fund the recovery effort — when necessary.

"As the federal government meets its responsibilities, the people of the Gulf Coast must also recognize its limitations," Bush said. "The engine that drives growth and job creation in America is the private sector, and the private sector will be the engine that drives the recovery of the Gulf Coast."

Entergy New Orleans heard the siren call of a bailout.

In November of 2005, according to a Reuters nuclear industry, including nearly $6 billion in operating tax credits, over a billion in decommissioning subsidies for aging plants and a 20 year extension of liability caps for accidents at nuclear plants. $3 billion in research subsidies and a nearly equivalent sum for construction subsidies was included for new plants, an effort that put Entergy out in the front of the pack. Entergy Corp. made bipartisan campaign contributions totaling $1,205,983 to federal candidates during the 2004 election cycle.

Gary Tayler, CEO of Entergy Nuclear, Inc. said on May 25, 2005 at a meeting of the Nuclear Energy Institute in Washington, DC, that Entergy is enjoying a "full partnership" with the Department of Energy in its quest to pave the way for the next generation of nuclear reactors.

"We're putting in money and they're matching it on a dollar for dollar basis," Taylor is quoted in the July/August 2005 issue of Nuclear Plant Journal. "Clearly, they have a responsibility to ensure that this is going in the right direction." He also said that the "President is trying to say that we [the government] will provide some financial certainty if things fall apart..."

On August 7, 2002, Synapse Energy Economics published a report, "Financial Insecurity: The Increasing Use of Limited Liability Companies and Multi-Tiered Holding Companies to Own Nuclear Power Plants," which was commissioned in part by Riverkeeper.

Former Commissioner of the NRC, Peter Bradford, wrote in his the foreword that the Synapse report "dissects a troublesome set of developments on the cusp between economic and safety regulation, namely, the arrangement of nuclear plant ownership into the limited liability subsidiaries of a few large companies. Because this arrangement has occurred during a period of lax and dispirited regulation, some important issues have not

report, Entergy unveiled a $3 billion plan that it said would "ensure liquidity" while it "awaits recovery." During a conference call with analysts at that time, Leonard said the company intended to be "relentless in recovery of storm costs."

"New Orleans is Entergy's home and we are absolutely dedicated to the city's reconstruction and resurrection," said Leonard immediately after the storm. "We are hopeful that we will be able to return home soon. Our ability to do that depends, of course, on a number of factors over which we do not have complete control."

"We are heartsick at the losses our communities and employees have suffered," said Curt L. Hebert, executive vice president, external affairs at Entergy Corp. in a public statement. "Even as we launch the largest power restoration in our country's history, we are equally concerned about reaching out to help our co-workers, families and neighbors restore their lives. Together we can and will rebuild and put this storm behind us."

Hebert is a product of the corporate-political revolving door. He was appointed chairman of FERC in 2001 by Bush, but stepped down months later, in September 2001, to take his position at Entergy Corp. He was in charge of lobbying the federal government for aid money after Katrina. That put him at odds with the chairman of the Gulf Coast Recovery and Rebuilding Council, Allan B. Hubbard, whose job it was to explain to Entergy that the feds would not be underwriting the company's New Orleans reconstruction effort.

In a letter dated November 16, 2005, Hebert noted he was "gravely disappointed," that the people of New Orleans would "suffer significantly" as a result of the Bush Administration's "fundamentally flawed" perspective. This stance, he charged, "repudiates the promise" made by Bush in Jackson Square.

been pursued effectively."

Years of "reckless undermining," he said, have now been "exposed in a series of financial collapses," among them a "ruinous mix of money, pressure...complexity and ideology."

"Regulating in this way is like driving drunk," Bradford went on. "Taxpayers, utility customers and power plant neighbors who thought themselves protected by firm requirements may one day wear the stunned expressions of Enron retirement plan holders or WorldCom investors."

According to Synapse, limited liability structure is an "effective mechanism" for transferring profits up the chain while creating a shield for the parent in case an unanticipated cost occurs at one of the plants.

"I also want to state clearly," Hebert wrote, "that [Entergy's shareholders and bondholders] have invested in a regulatory agency with the knowledge that the government regulates and protects their opportunity to earn a return."

With those words, Hebert inadvertently laid out the billion-dollar question: where do the boundaries of corporate welfare start and end?

In a response dated November 18, Hubbard pointed out Entergy's healthy bottom line while musing about the "inappropriate" nature of asking the federal government a handout while clasping a fistful of dollars.

"You told us that the ... board of the parent has a fiduciary duty not to take funds from other subsidiaries and use them to subsidize New Orleans," Hubbard wrote. "We respect the right of your board to decide how to allocate financial resources, such as last year's $909 million in earnings among various parts of Entergy Corp. We in turn believe it is inappropriate to transfer taxpayer resources to those investors after the fact for a risk they chose to take."

Prudent investors, he added, consider the risks inherent in any investment they make, including the risks of a natural disaster.

Ten days later, in a seven-page letter that smacked of one-upmanship, Hebert threatened that without "immediate federal assistance, it is unlikely that Entergy New Orleans can continue as a viable commercial entity." The threat was on the table: pay up or we pull up stakes.

Hebert also took exception to Hubbard's analysis, arguing that investment risk in a private company might be one thing, but risk in a publicly regulated utility quite another. He insisted that such public-private companies are by nature "entitled the opportunity to recover ... storm restoration costs."

Hebert says the obvious parallel and precedent was September 11. In the aftermath of that disaster, Congress passed the 2002 Supplemental Appropriation Act for Further Recovery From and Response to Terrorist Attacks on the United States. The $783 million in resulting CDBG funding included restoration of utility infrastructure for ConEdison, which, like Entergy New Orleans, is a publicly regulated utility and a subsidiary of a vast holding company. The airline industry, also, was offered a $5 billion bailout after 9/11 and a guarantee of up to $10 billion.

What Can Be Predicted Can Be Insured Against

But is a terrorist attack really a precedent for a natural disaster?

Not at all, said Howard Kunreuther, Cecilia Yen Koo Professor of Decision Sciences and Public

Policy and Co-Director of the Wharton Risk Management and Decision Processes Center at the University of Pennsylvania's Wharton School. Kunreuther has studied and written about the issue of insurance extensively, including an August 2005 report on "Terrorism Risk Financing in the United States," which outlines the many ways in which terrorism is a unique threat, and a January 2006 University of Pennsylvania Press publication, Has the Time Come for Comprehensive Natural Disaster Insurance? This work includes a chapter, "On Risk and Disaster: Lessons Learned from Hurricane Katrina ."

"9/11 is not a fair precedent for a natural disaster at all," Kunreuther told CorpWatch. "They're both very different in how they've been treated by the insurance industry. With a terrorist attack, you have less control. With a natural disaster, you can protect yourself to some degree."

Those who claim that Katrina was a completely unpredictable event may not have seen the many articles in publications ranging from National Geographic to a five-part series published pre-Katrina in the New Orleans Times-Picayune detailing the potential ramifications of just such a storm. And then there was Eric Berger's article in the Houston Chronicle on December 1, 2001, months after the Bush Administration's announcement of intent to downsize FEMA, in which it was reported that FEMA had declared the top three likeliest devastating emergencies in the United States to be a terrorist attack in New York City, a hurricane hitting New Orleans and a massive earthquake on the San Andreas fault.

Deregulation Comes Back to Roost

According to a May 2004 report from the United States General Accounting Office (GAO), limited liability companies such as Entergy Corp resulted from the deregulation of the electricity industry in the 1990s. "Like a partnership," the report said, "the profits are passed through and taxable to the owners ... like a corporation, it is a separate and distinct legal entity and the owners are insulated from personal liability for its debts and liabilities."

Such structures are made of loopholes the way some castles are made of sand—both, it turns out, can crumble under the sheer force of water.

"Entergy is a great example of how a company can shift liabilities to maximize profits while limiting liability," said Phillip Musegaas, senior policy analyst for the environmental watchdog group Riverkeeper, which is fighting to shut down the Entergy-owned and operated Indian Point nuclear power plants in Buchanan, New York. "Corporate restructuring is very sophisticated. They know their way out of regulation. They are way ahead of us."

All Profit, No Risk

It isn't "fair," said Entergy spokesman Stewart, to pass the cost of reconstructing Entergy New Orleans, the smallest subsidiary under the Entergy Corp umbrella, along to shareholders when the future fate of the company is still uncertain. Stewart explained that each subsidiary is a "separate business," and that each company is "protected from the burden" of picking up unexpected costs from the others.

"It would be irresponsible," Stewart told us. "It would not be prudent to invest shareholder money into the utility if there's no chance of recouping the money."

The corporation also has a "moral responsibility," he added, not to "risk the retirement funds of employees when we don't know if customers are returning."

Utility investors rely on a "business model," he said, that allows for the utility to pass the costs of reconstruction and damage from natural events along to ratepayers. With the exception of totally decimated areas of New Orleans, power has been returned to "anyone who can take it." He likened the devastation in the region to that suffered by orange juice producers when frost nips the growing season short.

"The price of juice goes up," he said, "and that's the models investors have invested in here."

Elizabeth Raley, spokesperson for Entergy New Orleans, acknowledged that customers are "very upset."

"We're faced with many customers who have extremely high bills because the natural gas wells in the Gulf of Mexico were damaged," she said. "The market rate for gas is passed on to our customers. ... We're working with customers to help them find a way to pay their bills. It's challenging to serve our customers, but that's what we're doing."

New Orleanians are wary and suspicious of what Entergy may do next. "Our customers have been through life-altering situations," Raley said. "I wouldn't be surprised if some of them feel hostility simply because they are having a rough time."

It turns out that the hostility dates from long before the storm, evidenced by a series of local and federal lawsuits. In an SEC 10-K filing, Entergy Corp describes its relationship with the state of Louisiana as "particularly litigious." 480 class action lawsuits including 10,000 claims have been filed against Entergy's various Gulf Coast subsidiaries, including Entergy New Orleans, for damages allegedly caused by disposal of hazardous waste and "asbestos-related disease" by contractor employees who worked between the years 1950 and 1980 and claim to have been exposed to hazardous materials during that time. In March 2004, endangered brown pelicans were found dead near Entergy New Orleans' Michoud power plant intake structure and return trough. And then there were the ratepayer lawsuits brought on by the City Council, during which it was alleged in testimony filed over a period of years that customers had been overcharged by upwards of $100 million. When the matter was settled in April 2003, ratepayers were reimbursed $11.3 million when it was found that Entergy had been incorrectly calculating the cost of fuel and passing the error along to customers.

This had all been sorted out since, according to Clinton A. Vince, managing partner of Sullivan & Worcester's Washington DC office, which represents the New Orleans City Council, the regulatory agency responsible for oversight of Entergy New Orleans. Vince supports the idea of a CDBG bailout.

"I won't second-guess them on their system. Nobody ever could have anticipated a flood of this magnitude," he said when asked why pipes that were susceptible to corrosion salt water were buried underground in a city that lies below sea-level. "This was not a reasonably foreseeable event."

"Entergy needs a reorganization plan," Vince said. "The content of that plan will change drastically based on CDBG funds. There would be dire consequences if Entergy walked away. We need the utility to stay there and rebuild the system. It's a unique situation—the worst disaster for a utility in the history of the country. People have lost everything, and it would be unrealistic and unfair to pass those costs along [to ratepayers]. People would have no incentive to return."

* * *

On March 30, 2006, Donald Powell, the Bush administration's Gulf Coast recovery coordinator, said the revival of the Crescent City could take up to a quarter of a century and also hinges on factors that are "out of our control." The amount of total funds that will be allotted by the state and federal governments, Powell said, is still up in the air.

Will Entergy choose – or be forced – to cut and run from New Orleans?

"Our plan is to stick it out," Entergy New Orleans' spokeswoman Raley told Corpwatch. "We're very hopeful that our funds will come through, and we'll be able to continue to operate."

EXHIBIT X

April 25, 2003

MEMORANDUM TO: Chairman Diaz

FROM: Hubert T. Bell
Inspector General /RA/

SUBJECT: NRC ENFORCEMENT OF REGULATORY REQUIREMENTS AND COMMITMENTS AT INDIAN POINT, UNIT 2 (CASE NO. 01-01S)

Attached is an Office of the Inspector General (OIG), U.S. Nuclear Regulatory Commission (NRC) Event Inquiry that addresses the NRC's oversight of operations at the Indian Point, Unit 2 nuclear power plant in Buchanan, New York.

Please call me if you have any questions regarding this Event Inquiry. This report is furnished for whatever action you deem appropriate. Please notify this office within 90 days of what action, if any, you take based on the results of the Event Inquiry.

Attachment: As stated

cc w/attachment:
Commissioner Dicus
Commissioner McGaffigan
Commissioner Merrifield
W. Travers, EDO

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04/ **25** /03	04/ **24** /03	04/ **25** /03	04/ **25** /03	04/ **25** /03

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OFFICE OF THE INSPECTOR GENERAL
EVENT INQUIRY

UNITED STATES NUCLEAR REGULATORY COMMISSION

NRC ENFORCEMENT OF REGULATORY REQUIREMENTS AND COMMITMENTS AT INDIAN POINT, UNIT 2

Case No. 01-01S

/RA/
Veronica O. Bucci, Special Agent

/RA/
George A. Mulley, Jr.,
Senior Level Assistant for Investigative Operations

/RA/
Brian C. Dwyer
Assistant Inspector General for Investigations

NRC ENFORCEMENT OF REGULATORY
REQUIREMENTS AND COMMITMENTS
AT INDIAN POINT, UNIT 2

Case No. 01-01S April 25, 2003

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BASIS AND SCOPE

The Office of the Inspector General (OIG) initiated this inquiry in response to a Congressional request that OIG examine issues concerning U.S. Nuclear Regulatory Commission (NRC) oversight of operations at the Indian Point 2 (IP2) nuclear power facility in Buchanan, New York. The request referred specifically to "internal Con Ed/Indian Point 2 condition reports" made public in a January 2001 petition review board meeting that "may include information which indicates that the plant operator may be in violation of a commitment made back in 1997 regarding design bases requirements."

The Congressional request also focused on issues raised by an engineering consultant hired by the licensee who had recently resigned his position due to a differing professional opinion regarding the plant's Reactor Protection System. The request noted that one of the more lengthy condition reports cited discrepancies between design drawings and the as-built configuration of the Reactor Protection System.

Based on the above concerns, OIG initiated an Event Inquiry to examine:

> I. NRC's oversight of IP2's progress toward fulfilling two design bases commitments made to the NRC in 1997. These commitments were made in response to NRC's 1996 request for information concerning plant programs and processes for controlling and maintaining operations within the facility's design bases.
>
> II. NRC's response to the specific concerns raised by an IP2 engineering consultant pertaining to discrepancies between design drawings and the as-built configuration of the Reactor Protection System.
>
> III. NRC's oversight of IP2's corrective action program between 1995 and 2001.
>
> IV. NRC's utilization of its Senior Management Meeting process to heighten attention to IP2.

BACKGROUND

NRC's Regulation of Power Plants — Overview of Terms Used in This Report

Nuclear power plants are required to adhere to U.S. Nuclear Regulatory Commission (NRC) regulations to ensure their safe operation. These regulations include requirements that power plants operate in accordance with their current license, which includes (1) the plant's technical specifications, (2) license conditions, (3) licensee commitments made in response to NRC Generic Letters and Bulletins, and (4) the Final Safety Analysis Report (FSAR).[1] Design bases information identifies the specific functions to be performed by a power plant's structures, systems, and components as well as associated design parameters.

In addition, plants are required to have a corrective action program (CAP) that enables them to identify, prioritize, and correct problems in a timely manner. Power plants manage their CAP by maintaining a database of action items, or condition reports, which describe particular plant conditions in need of repair or attention. Plants typically prioritize these condition reports based on safety significance and address them accordingly.

NRC provides oversight of nuclear power plants to ensure that plants are operating safely. The agency conducts reactor inspections to determine whether power plants are in compliance with agency requirements. Inspections range from routine, baseline inspections[2] to inspections beyond the baseline which may focus on areas of declining plant performance. The agency issues sanctions (i.e., enforcement actions) — such as Notices of Violation (NOV),[3] fines, or orders to modify, suspend, or revoke licenses — when plants are out of compliance. In 2000, NRC implemented a Reactor Oversight Process (ROP), which was intended to be substantially different from the previous oversight process and to take into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and evaluating the safety performance of NRC-licensed plants. Under this process, inspection findings are evaluated for risk significance using pre-established criteria. Plants that fail to meet certain safety objectives, as determined by performance indicators and inspection findings, are to receive increased inspection activity, focusing on areas of declining performance and may be subject to enforcement action.

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[1]The FSAR is a licensing document that provides a description and safety analysis of the site, the design, design bases and operational limits, normal and emergency operation, potential accidents, predicted consequences of such accidents, and the means proposed to prevent or mitigate the consequences of such accidents. When the FSAR has been updated, it is referred to as the updated FSAR, or UFSAR.

[2]Baseline inspections are common to all nuclear power plants; NRC's baseline inspection program is the normal inspection program performed at all nuclear power plants. The program focuses on plant activities that are "risk significant," that is, those activities and systems that have a potential to trigger an accident, can mitigate the effects of an accident, or increase the consequences of a possible accident.

[3]An NOV formalizes a violation by identifying a requirement and how it was violated.

Between 1986 and 2001, NRC also used its semiannual Senior Managers Meetings (SMM)[4] as a means to increase attention to plants with persistent operational problems. During these meetings, the agency's senior managers reviewed certain plants experiencing declines in performance. Participants decided whether to increase oversight of subject plants and, if so, by what means. For example, a SMM decision might require a plant to undergo additional inspections, or the staff could issue a "trending letter" to advise a licensee that NRC had taken note of declining plant performance, or designate the plant as in need of heightened NRC attention (e.g., designation as an Agency Focus Plant).

One way in which nuclear power plants fulfill NRC expectations is through regulatory commitments. Regulatory commitments are non-binding statements made by licensees to NRC indicating they will take specific actions, for example, to verify the accuracy of UFSAR information, and they typically reflect the means by which licensees will accomplish the commitment (e.g., in a certain timeframe, following a specific approach).

The Indian Point Nuclear Power Plant, Unit 2 (IP2), is one of two operating pressurized water reactors located in Buchanan, NY, 24 miles north of New York City. IP2 began commercial operations in August 1974. The Consolidated Edison Company of New York, Inc. (ConEd), owned IP2 until September 6, 2001, when the plant was purchased by Entergy Nuclear Operations, Inc. NRC's Region I office[5] provides oversight for IP2.

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[4]The Senior Management Meeting (SMM) program which required semiannual meetings of NRC senior managers was replaced in 2001 by the Agency Action Review Meeting (AARM) program. The AARM is an annual meeting of NRC senior managers under the Reactor Oversight Process. This meeting essentially replaces the SMM under NRC's previous oversight process.

[5]NRC has four regional offices that conduct inspections of nuclear reactors within regional boundaries. NRC's Region I provides regulatory oversight for IP2 and other nuclear facilities within the northeast region of the United States.

DETAILS

I. NRC OVERSIGHT OF IP2'S PROGRESS TOWARD FULFILLING TWO 1997 DESIGN BASES COMMITMENTS

Overview of Design Bases

Nuclear power plants are designed so that internal and external events (e.g., loss of coolant accident, fire, earthquake) will not jeopardize plant safety or threaten the health and safety of the public. A plant's design bases in part describe how the plant will cope with various accidents and emergencies. Plant structures, systems, and components (SSC) must be built in accordance with design requirements that will enable the plant to meet its design bases and, consequently, to withstand such accidents and emergencies. Plant operators are expected to not make plant modifications to safety related systems without having performed NRC required safety analyses, which are needed to prove the modification will not affect the plant's ability to meet its design bases requirements. Furthermore, when modifications are made, they are supposed to be reflected in the plant's design bases documents, which link each plant SSC to its design bases and original design requirements. Design bases documents include such information as industry, regulatory, and manufacturer criteria for plant systems and information generally contained in the UFSAR specifying system functions and requirements, component functions and requirements, interface requirements from supporting and supported systems, applicable accident analysis assumptions related to the systems, and plant design drawings and calculations.

NRC Requests Licensee Feedback on Design Bases Issues

NRC team inspections during 1995 and 1996 identified concerns regarding the ability of NRC licensees to maintain and implement the design bases at certain plants. To learn more about the scope and extent of the problems among operating nuclear power reactors, the staff proposed that all licensees be required to provide information regarding the availability and adequacy of design bases information. To that end, on October 9, 1996, NRC issued a letter to each NRC reactor licensee in accordance with Title 10, Part 50, Section 54(f), Code of Federal Regulations (10 CFR 50.54(f)) requesting that each licensee submit under oath a written response within 120 days describing and discussing the effectiveness of its programs and processes for controlling and maintaining operations within the facility's design bases. The stated purpose of the letter was "to require information that will provide the U.S. Nuclear Regulatory Commission (NRC) added confidence and assurance that [licensee plants] are operated and maintained within the design bases and any deviations are reconciled in a timely manner."

Specifically, NRC found it problematic that some licensees had failed to (1) appropriately maintain or adhere to plant design bases, (2) appropriately maintain or adhere to the plant licensing basis, (3) comply with the terms and conditions of licenses and NRC regulations, and (4) assure that the UFSARs properly reflect the facilities. According to the letter, "The extent of the licensees' failures to maintain control and to identify and correct the failures in a timely manner is of concern because of the potential impact on public health and safety should safety systems not respond to challenges from off-normal and accident conditions."

NRC Reviews Overall Response

Subsequent to NRC's receipt and review of all licensee responses to the October 9, 1996, letter, the staff issued SECY-97-160,[6] which described a four-phased approach which NRC had undertaken to review the licensee responses to the 10 CFR 50.54(f) request. The SECY described the completion of the first three phases and concluded that all licensees had established programs and procedures to maintain the design bases of their facilities. However, SECY-97-160 also recommended certain plant-specific, final-phase followup activities to address the staff's concerns about either (1) the performance of certain licensees in controlling facility design bases or (2) the need to validate the effectiveness of a particular element of a licensee's design control program.

A manager in the NRC's Office of Nuclear Reactor Regulation (NRR) told OIG that the request began with a high level of agency concern that there were widespread problems pertaining to the accuracy of plant UFSARs and there was a heightened awareness that these problems needed to be resolved as quickly as possible. However, as licensee efforts to address these concerns unfolded, NRC staff recognized that this effort was more resource intensive than had initially been anticipated, and staff allowed licensees to have more time to complete these efforts.

IP2 Responds to NRC Design Bases Request

In response to NRC's October 1996 10 CFR 50.54(f) request to ConEd regarding IP2, the licensee made two specific commitments. In its February 13, 1997, letter that conveyed these commitments to NRC, ConEd stated its intent "to voluntarily initiate and complete" an UFSAR review program. The program was scheduled for completion within 24 months. The UFSAR review program was to include (1) verification of the accuracy of the UFSAR design bases information, (2) assessment to confirm that the UFSAR design bases information was properly reflected in plant operation, maintenance, and test procedures, (3) review of the UFSAR to identify and resolve any internal disagreements or inconsistencies which could impact the design bases, and (4) development of a process to enhance overall the UFSAR accessibility. In its second commitment, ConEd stated it would continue its "Design Basis Document (DBD) Initiative" to review and update existing design bases documents and create new ones if needed. The continuation of the DBD Initiative was to include supplementation of 22 DBDs with a combination of additional DBDs and added information on interfacing systems. This effort was also to be completed in 24 months.

IP2 Extends Completion Date

In a letter dated February 17, 1999 (24 months after the initial commitments were made), ConEd provided an update to NRC concerning the commitments it had made pursuant to NRC's 1996 request. The letter reported that both the UFSAR verification effort and DBD initiative were underway; the UFSAR effort was approximately 65 percent complete and the

[6]SECY 97-160, "Staff Review of Licensee Responses to the 10 CFR 50.54(f) Request Regarding the Adequacy and Availability of Design Bases Information," dated July 24, 1997.

supplementation of 6 of 27 DBDs was in progress. The letter also changed the completion date of both commitments: December 31, 1999, for the former and December 31, 2002, for the latter.

OIG learned that NRC is not expected to formally approve changes in commitment completion dates such as the one described above. According to the NRR manager, commitments are often schedule or process related (e.g., licensee commitment to fix something by a specific time or in a particular manner) and changes in completion dates are not necessarily problematic. For example, the manager said, a rule may say to fix something in a timely manner and the licensee will commit to do so within 2 months. However, if the licensee fails to make the 2-month deadline, the licensee may adjust the timeframe to another date that NRC would consider timely.

The NRR manager and Region I staff told OIG that after IP2 became involved in these efforts, all parties realized that the 2-year timeframe that ConEd initially committed to was unrealistic. A number of plants, including IP2, required additional time to complete their review and NRC staff generally viewed these extensions as reasonable.

OIG also learned that with regard to ConEd's schedule change for the UFSAR commitment, Region I staff felt IP2's progress toward fulfilling the commitment was proceeding in a timely manner and that the schedule change was reasonable.

In June 2000, ConEd provided NRC with a new projected completion date of March 31, 2001, for its commitment to verify the accuracy of the UFSAR, and ConEd reported that it still anticipated completing its DBD initiative by December 31, 2002.

On December 31, 2002, Entergy forwarded correspondence to NRC modifying the completion date for the original commitment that was due on December 31, 2002, to a revised commitment date of December 31, 2003. According to the Region I Administrator and staff, the modification of the completion date was reasonable and acceptable. The Region I Administrator said he considered these deferrals to be appropriate given that numerous, more significant operational and design-related issues emerged over this period requiring extensive licensee management attention and resources.

Region I Oversees IP2 Progress in Fulfilling Design Bases Commitments

According to a Region I Branch Chief, he visited IP2 on April 3, 2001, and verified for himself that the UFSAR update was "essentially done" and that ConEd was "just wrapping up loose ends." The Branch Chief drew this conclusion based on a presentation ConEd gave him describing the methodology for and status of the UFSAR effort. Additionally, he stated that his conclusion was supported by a series of NRC inspections conducted at IP2 since the initial commitment that confirmed progress was being made. OIG reviewed NRC inspection reports from October 1997 through August 2002 and found that some of the NRC inspections specifically looked at the UFSAR and DBD efforts through baseline and special inspections. These reports reflected inspectors' observations that progress continued to be made in these efforts.

The Branch Chief also explained to OIG that when Entergy took over as the licensee for IP2 in September 2001, it assumed ConEd's commitment to complete its DBD Initiative by December 31, 2002, without modifying the completion date. Entergy incorporated the commitment into its "Fundamentals and Improvement Plan" for IP2. With regard to the status of the DBD commitment, the Branch Chief said he visited the plant in May 2002 at which time the plant had completed the review of 22 of the 27 DBDs and planned to complete 3 more by the end of 2002.

According to NRC Inspection Report No. 05-247/2002-010, dated August 28, 2002, which reported results of a supplemental and problem identification and resolution (PI & R) inspection from June 17 through July 19, 2002, Entergy had revised its schedule for completing the DBD effort. According to the inspection report, two remaining DBDs (fire protection and electrical separation) would be completed in 2003, rather than by December 2002. The inspection team concluded that the schedule modification was reasonable.

OIG FINDING

In February 1997, ConEd responded to NRC's 10 CFR 50.54(f) request for information by committing to two separate 24-month efforts at IP2. In the first of these two commitments to NRC, ConEd stated its intent to initiate and complete an UFSAR review program and in its second commitment, ConEd stated it would continue its IP2 DBD Initiative to review and update existing design basis documents and create new ones if needed. Although ConEd initially committed to complete both efforts in 2 years, ConEd revised its projected completion dates two times for the first effort. The UFSAR review program, initially expected to be completed by February 1999, was extended to December 1999, and finally completed by April 2001. The completion date for the second effort was also revised twice, once by ConEd and the second time by Entergy Nuclear Operations, Inc., IP2's current license holder. The DBD Initiative, initially slated for completion by February 1999, was extended to December 2002, and is now expected to be finished by December 31, 2003. OIG found that the NRC staff did not object to the time extensions because it believed each extension was reasonable, given other significant operational problems at the plant, the effort that was required to fulfill the commitments, and the licensee's steady, but slow, progress in addressing them.

II. NRC'S RESPONSE TO REPORTED DISCREPANCIES BETWEEN RPS DESIGN DRAWINGS AND AS-BUILT CONFIGURATION

The Reactor Protection System

The Reactor Protection System (RPS), a system described by NRC staff as "very safety significant" to nuclear power plant operations, is designed to detect a problem in the plant and, if the problem is serious enough, cause the plant to trip (i.e., to automatically shut down in an emergency situation). According to NRC staff, the system can be manually or automatically activated to initiate a plant shutdown. Staff said that to ensure that the reactor will shut down when necessary, the RPS features multiple, independent equipment and components. Any individual RPS component, therefore, could be significant. Furthermore, RPS interfaces with many other safety systems for process monitoring of safety parameters such as reactor coolant pressure, temperature and flow, pressurizer level, steam generator level, and reactor building pressure. As a result, staff said, deficiencies in other systems could have an effect on RPS's ability to operate during an event.

The Region I Administrator told OIG it is a significant problem if the as-built configuration of a system, such as the RPS, is inconsistent with what is needed for the system to be functional. He said it is of lesser significance, but still important, when a system's as-built configuration is inconsistent with design drawings but is still functional. He explained that in either case, inconsistencies between system configurations and design drawings may be indicators that other issues within the system warrant attention.

IP2 Condition Reports Identify Design Bases Discrepancies

OIG learned that in February 2001, a ConEd engineering consultant raised an allegation to Region I pertaining to design bases discrepancies between design drawings and the as-built configuration of the RPS. The allegation referred to 13 IP2 condition reports (CR) that IP2 plant personnel, including the engineering consultant, had written to describe these issues. These CRs were a subset of a larger number (more than 300) of CRs written on RPS between 1998 and 2001.[7] This subset of CRs identified circumstances in which the system's wiring violated statements in the UFSAR. For example, the CRs identified instances of wires associated with computer and alarm circuits being in close proximity of and sometimes in the same cable tray as the wires associated with the trip and logic circuits. The CR reported that these as-built wiring configurations were in conflict with UFSAR wiring separation criteria.

OIG reviewed summaries of the 13 CRs raised in the allegation. Eight of the 13 (CRs 199803574, 199902835, 199903445, 199904968, 200007597, 200009499, 200009641, 200010125) focused on:

- Quality assurance requirements for design verifications,
- Wiring changes resulting from modifications that could not be located, and
- Wiring configurations not in accordance with UFSAR separation requirements.

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[7]As context, both regional staff and the IP2 engineering consultant told OIG that roughly 10,000 CRs were being written per year during this timeframe concerning IP2 conditions perceived by licensee staff as in need of attention.

A ninth condition report (CR 200100327) summarized the eight preceding CRs. The remaining four condition reports (CRs 199900478, 199902274, 200008415, and 200008818) documented additional examples of related RPS wiring discrepancies. (See Appendix A for a listing of the 13 CRs and a description of the issues covered in each.)

The engineering consultant told OIG that while employed at IP2 he wrote CR 200100327 as a summary after becoming aware of the eight earlier CRs. These eight CRs summarized documented deficiencies such as wiring separation issues, wiring configurations not in accordance with design drawings, and cable splices not identified on drawings.

The engineering consultant told OIG that he was concerned that collectively these issues warranted a higher level of attention than ConEd had determined was appropriate and that he had raised the matter with ConEd management. Specifically, he explained, he wanted ConEd to perform another Operability Determination (OD) on the RPS to determine whether the system in its current configuration was operable. He told OIG that prior to his writing of CR 200100327, ConEd performed an OD (OD 00-018) on RPS that addressed a subset of the issues raised in CR 200100327. However, he explained that in his opinion that OD did not go far enough to assess the functional changes that may have resulted from the as-found wiring conditions. Dissatisfied with ConEd's response to the issues he raised, and concerned that ConEd would downgrade CR 200100327 from Significance Level (SL) 2 to an SL3,[8] the engineering consultant formally raised the matter to Region I as an allegation.

NRC's Response to RPS Design Bases Discrepancies

OIG reviewed documentation of NRC's response to the issues raised by the engineering consultant and learned that NRC:

(1) inspected several RPS deficiencies prior to the engineering consultant's allegation,

(2) conducted an inspection focused specifically on the RPS wiring discrepancies described in CR 200100327, and

(3) responded directly, in writing, to the engineering consultant on the outcome of NRC's review of the concerns he raised in his allegation.

In the following three sections, OIG describes each of these efforts, which OIG learned, collectively addressed each of the 13 CRs mentioned in the engineering consultant's allegation.

(1) NRC Inspects RPS Deficiencies

OIG learned that prior to receipt of the allegation from the ConEd engineering consultant, and during the course of escalated regulatory activities by Region I subsequent to a steam

[8]IP2 CRs were ranked on a scale of 1 through 4, with SL1 assigned the highest level of significance. The engineering consultant explained to OIG that CRs assigned a higher SL would receive a more heightened response from ConEd. For example, CRs assigned as SL2 were required to receive a formal Operability Determination, while this was not a requirement for CRs assigned as SL3.

generator tube rupture that occurred at IP2 in February 2000,[9] a team of Region I inspectors conducted a 7-week inspection of "engineering, operations and maintenance, radiation protection, security, and weld radiographs associated with the steam generator replacement project." Inspection activities included a review of a sample of RPS open corrective action items relating to the RPS's nonconformance with design drawings and the UFSAR.

OIG reviewed the inspection report findings pertaining to the RPS review. The inspection report (IR 05-247/2000-014), dated January 2001, described the RPS issue as follows:

> The issue involved the licensee's observation that wiring within the protection racks did not always conform with the statements contained in the UFSAR and electrical separation criteria contained in drawing A208685. Specifically, the licensee found instances of wires associated with computer and alarm circuits being in close proximity of, and sometimes in the same cable tray as, the wires associated with the trip and logic circuits. The licensee also identified examples of switch contacts originally reserved for logic and trip function being used for computer and alarm functions. All potential interactions involved a single train of protection logic and low energy and low voltage circuits.

According to the NRC inspection report, the inspector reviewed three CRs mentioned in the engineering consultant's allegation (CRs 200007597, 200008818, and 200009499) related to RPS logic rack wiring separation concerns, OD 00-018 (dated November 28, 2000), and OD supporting documentation. Based on this review, the report concluded, "There were no significant findings associated with this issue."

The Region I inspector who conducted the review told OIG that the inspection was focused on ensuring that the discrepant conditions reported in the three CRs did not affect the safe operation of the RPS. Although the inspector acknowledged to OIG that it was better to review all open issues and CRs related to a particular system and to sample closed CRs, the inspector explained that he did not do so because of the limited scope of the review coupled with limited manpower resources and time. The Region I Administrator explained to OIG that this sampling of RPS issues was part of a larger review of deficiencies and corrective actions that needed to be addressed at the plant.

(2) NRC Inspects RPS Wiring Discrepancy Issues Described in CR 200100327

OIG learned that following the engineering consultant's allegation pertaining to the RPS, NRC inspectors revisited the issues that the consultant had collectively recorded in CR 200100327 and documented their findings in a June 2001 inspection report (IR 05-247/2001-005) which described the Region I inspectors' review of:

> Corrective actions taken by ConEd to address issues raised in CR 200100327;

[9]On February 15, 2000, IP2 experienced a steam generator tube rupture in one of the plant's four steam generators, which resulted in a minor radiological discharge to the atmosphere.

ConEd's February 12, 2001, OD 01-002, "Ensuring the Functional Capability of a System (RPS) or Component," to determine whether the bases used in the OD were valid and accurate;

Safety Evaluation 99-160-EV to change the UFSAR such that wire separation between safety and non-safety wires was no longer required; and

RPS open condition reports.

These inspection efforts are described below.

Corrective Actions Taken to Address CR 200100327 Issues

OIG reviewed IR 05-247/2001-005, which described Region I's examination of the licensee's corrective actions associated with CR 200100327 and the eight feeder CRs, and corrective actions pertaining to CR 200008415 and one additional CR not referenced in the allegation. The inspection report indicated that as background for the inspection, NRC reviewed CRs 199900478 and 199902274, which had been referenced in the allegation. According to the inspection report, inspectors also:

- Reviewed a ConEd evaluation titled, "SL-2 Evaluation for CR 200100327 on the Reactor Protection System," dated March 7, 2001, to confirm that this evaluation addressed appropriate root causes, contributing causes, compensatory actions and the proposed corrective actions.

- Attended a Corrective Action Review Board (CARB) meeting which reviewed and discussed the evaluation.

- Reviewed the list of ICA (Implementation of Corrective Actions) for CR 200100327 to confirm that the listed corrective actions adequately addressed the root causes and the concerns raised in CR 200100327.

- Reviewed a sample of corrective actions and issues to determine whether these corrective actions were timely and appropriate to address the issues.

- Reviewed the rationale provided for delayed corrective actions.

- Reviewed IP2 documents to confirm that on February 12, 2001, ConEd had generated OD 01-002, "Ensuring the Functional Capability of a System (RPS) or Component," to demonstrate that the RPS can perform its safety function, in spite of the combined wiring and documentation deficiencies.

- Reviewed IP2 documents to confirm that on March 12, 2001, ConEd completed a safety evaluation to address the wiring separation issue regarding RPS wiring configuration conformance with the UFSAR.

Based on this review, the inspectors found no issues that would render the RPS incapable of performing its intended safety function. Specifically, the inspection report stated that no findings of significance were identified.

ConEd's Operability Determination (OD) 01-002, "Ensuring the Functional Capability of a System (RPS) or Component"

According to IR 50-247/2001-005, ConEd generated OD 01-002 "to demonstrate that the RPS could perform its safety function." OIG learned that Region I inspectors reviewed OD 01-002 to determine whether the bases used in the determination were valid and accurate. The inspectors also reviewed supporting documents used in the OD to verify that the data and bases were accurately translated. Supporting documents reviewed included RPS test procedures and test results, a modification for replacing 88 relays in the RPS, and a sample of CRs associated with RPS wiring issues. CRs reviewed included CR 200008818 and two additional CRs not mentioned by the alleger. Based on their review of this issue, the Region I inspectors again concluded that there were "no findings of significance."

Safety Evaluation 99-160-EV

Inspection report 50-247/2001-005 noted that in March 2001, ConEd generated a safety evaluation (SE 99-160-EV) to change the UFSAR so that wire separation between safety and non-safety wires would no longer be required and "safety and non-safety wires can run together within a panduit inside the RPS cabinet." However, according to the Region I inspectors, the safety evaluation did not provide sufficient rationale to justify the change to the UFSAR. According to the inspection report, this matter was not resolved during the inspection and was referred to NRR for review. OIG learned that the results of NRR's review were documented in IR 50-247/2001-010, dated December 17, 2001. In that inspection report, NRR acknowledged that SE 99-160-EV failed to address certain relevant issues; however, NRR concluded that the wiring separation between safety and non-safety wires inside the RPS cabinets was not a design requirement for IP2 and was in compliance with industry standards. Consequently, the wiring configuration at IP2 met design requirements and the issue was closed.

RPS Open Condition Reports

As part of this inspection effort, inspectors also reviewed the RPS condition report history since 1998 and found that since that time more than 300 CRs had been written on the RPS. As of March 9, 2001, 47 CRs remained open in the database, some for almost 3 years. ConEd's records indicated that of the 47 CRs, 3 were ranked as SL4; 37 were ranked as SL3; and 7 were ranked as SL2. The inspection report stated that in response to the inspectors' concerns about possible combined operability or functional effects from the 47 open CRs, ConEd performed an overall assessment of the 47 open CRs and concluded that no functional problems existed. The inspectors reviewed a sample of four CRs to confirm that there were no combined effects that could challenge the functionality of the RPS. The selected CRs were, based on the inspectors' judgement, most likely to yield inspection findings. Based on this review, the inspectors again identified no findings of significance.

(3) Region I Response to Engineering Consultant's Allegation

In a letter dated July 19, 2001, Region I formally responded to the ConEd engineering consultant who wrote CR 200100327 and who subsequently raised the RPS-related issues to Region I. The letter summarized the consultant's RPS-related concerns as presented in CR 200100327, relayed NRC's inspection findings (from IR 50-247/2001-005) pertaining to these concerns, and described the licensee's actions to address them. In its letter to the engineering consultant, Region I addressed the consultant's concern that "there is a lack of response effort and inadequate corrective actions in response to concerns [the consultant] raised regarding deficiencies in the design record and configuration control of the Reactor Protection System (RPS)." The Region I letter also addressed the consultant's concern that OD 00-018 "adequately addressed RPS wire separation and isolation issues, but not the broader concerns" (i.e., loss of design control due to wiring configurations). The letter explained that in response to these concerns, NRC completed an inspection of RPS wiring issues at IP2 on May 4, 2001, that was documented in IR 50-247/2001-005.

The letter also explained that to address the "broader issue for the RPS wiring," ConEd completed an RPS operability determination (OD 01-002) on February 12, 2001, completed a root cause evaluation for CR 200100327, entitled, "SL-2 Evaluation for CR 200100327 on the Reactor Protection System," on March 7, 2001, and established a corrective action program to address other broader aspects of the RPS wiring deficiencies.

In its conclusion to the consultant's concern about RPS configuration control/design record deficiencies, the letter stated,

> . . . your concern was partially substantiated. There were design control weaknesses at IP-2. However, at the time of our inspection, ConEd had established a corrective action plan to address the broader issue as described above [i.e., loss of design control]. Further, our inspection did not uncover any issues that would render the RPS incapable of performing its intended safety function.

The letter also addressed the consultant's concern that CR 200100327, initially assigned an SL of 2, would be reassigned an SL of 3 and that, as a result, ConEd would not conduct an OD "or otherwise address the broader operability issue raised by the lack of quality control in the changes made to the RPS." The letter explained that (1) the licensee did, in fact, complete an OD for the RPS (OD 01-002), which "addressed some important wiring issues;" (2) NRC's inspection did not identify any issues that would affect the functionality of the RPS; and (3) CR 200100327 remained as an SL2 CR.

The Region I inspectors responsible for reviewing the concerns identified by the engineering consultant told OIG that they did not find anything that would render the RPS inoperable.

<u>OIG FINDING</u>

Beginning as early as 1998, ConEd identified problems associated with the IP2 RPS wiring configurations and generated internal CRs to document the findings. These CRs identified circumstances in which the system's wiring violated statements in the UFSAR. Thirteen CRs identifying (or reiterating) such wiring discrepancies were presented formally to the NRC as an allegation by an IP2 engineering consultant who was concerned that collectively the RPS wiring discrepancies warranted a higher level of attention than the licensee had determined was appropriate. OIG learned that Region I performed two inspections relative to these issues and the NRC's Office of Nuclear Reactor Regulation documented its review in a third inspection report. In addition, Region I responded directly to the engineering consultant in a letter dated July 19, 2001. OIG determined that the NRC appropriately responded to the allegations presented to Region I by the engineering consultant. OIG's review of the three inspection reports and Region I's response to the engineering consultant determined that while NRC validated some of the issues the consultant had raised, the agency repeatedly concluded there were no "findings of significance" related to the RPS wiring issues and that ConEd had appropriate measures in place to address the conditions.

III. NRC REGULATORY OVERSIGHT OF IP2'S CAP: 1995 – 2001

Overview of IP2 Operational Problems

Between 1995 and 2001, IP2 experienced a series of operational problems, attributed in part to deficiencies in IP2's corrective action program (CAP) (i.e., its program to self-identify and resolve plant problems). For example,

- In 1995, plant personnel cleaned a turbine using grit. The grit caused significant damage to the internal components of a heater drain tank pump and migrated unchecked throughout the feedwater system, surfacing 2 years later and causing valves to operate erratically.

- NRC inspections conducted between 1996 and 1997 identified various issues, including weaknesses in corrective actions taken to address problems identified by the plant. As a result, in May 1997, NRC issued an NOV citing IP2 for nine violations of NRC requirements, six of which were attributed to corrective action violations.

- In the fall of 1997, IP2 voluntarily shut down to address a large backlog of equipment, programmatic, and performance problems. The plant remained out of service until September 1998.

- In 1998, in NRC Evaluation Team Report 05-247/1998-005, NRC noted that IP2 had identified problems with its CAP in that its corrective action processes were cumbersome and inefficient, many corrective actions were untimely, and completed actions were typically not revisited to determine whether they had achieved their goal.

- In August 1999, IP2 experienced a significant reactor trip, or shutdown, partly due to weaknesses in its CAP.

- In February 2000, IP2 experienced a steam generator tube rupture, also partly attributed to weaknesses in the plant's CAP.

- In May 2000, NRC categorized IP2 as an Agency Focus Plant, a status that denotes a need for increased oversight by NRC.

- In 2001, NRC found that IP2 continued to experience problems in its CAP, including issues pertaining to its RPS.

Significance of the Corrective Action Program

NRC inspects many aspects of nuclear power plants to ensure their safe operation, including the licensees' ability to identify and correct conditions that may affect plant performance and safety. Title 10 of the Code of Federal Regulations, Chapter 50 (10 CFR 50), Appendix B, directs licensees to have a program to assess problems in plant operations and to ensure that timely and effective corrective actions take place. Therefore, it is the licensee's responsibility to implement a program to identify and resolve problems at its facility. Historically this has been referred to as the nuclear power plant's CAP.

NRC Region I staff told OIG that overall plant performance is greatly determined by the effectiveness of a licensee's CAP. Staff told OIG that they expect licensees to be aggressive in identifying concerns and appropriately correcting problems, but they recognize that every plant has problems that need to be addressed. When a CAP is effective, staff said, a licensee is able to identify, prioritize, and quickly resolve conditions that may have a negative impact on plant operations. Staff said they have found that the better performing plants are very aggressive at correcting deficiencies. These plants are also proactive in conducting preventive maintenance and in monitoring plant equipment and conditions. As a result, staff said, those licensees have more durable solutions to their problems than poorer performing plants.

Several staff members interviewed by OIG observed a direct connection between ineffective CAPs and NRC's identification of a plant as an NRC Watch List[10] Plant. According to one staff member, in every case where a plant had problems or became an NRC Watch List Plant, there was a corresponding weakness in the licensee's ability to identify, evaluate, and correct problems, as well as a weakness in assessing the effectiveness of their corrective actions.

The Region I staff told OIG that if NRC lost confidence in a licensee's CAP, the agency would seriously consider whether the licensee should be permitted to operate.

NRC Identifies Repeated Problems With IP2 CAP

OIG was told by the Region I Administrator and staff that between 1995 and 2001, NRC dedicated significant resources to conduct inspections, document findings, and issue sanctions at IP2, yet problems persisted at the plant. Many of the inspections identified problems with IP2's CAP; however, despite heightened levels of NRC attention to these weaknesses, problems related to corrective actions remained unresolved. [See Appendix B for a chronology detailing the significant inspection activity and other oversight efforts performed at IP2 by NRC during this period.]

According to Region I staff, between April 1995 and February 2001, NRC conducted 20 special team inspections at IP2, logging 5,870 inspection hours dedicated to engineering and problem identification and resolution (PI&R).[11] By comparison, the average number of hours devoted to these types of inspections at other single unit[12] Region I nuclear power plants during the same period was 3,854. Furthermore, between 1995 and 2001, IP2 received 13 enforcement actions, 9 of which identified corrective action issues and 8 of which resulted in monetary fines. [See Appendix C for additional information on these 13 enforcement actions.] This expenditure of inspection resources at IP2 was NRC's response to a perceived downward performance trend

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[10]In 1999, there was a change in NRC terminology; Watch List plants are now referred to as Agency Focus Plants.

[11]NRC now refers to the CAP as problem identification and resolution (PI&R). This Event Inquiry, which covers a time period during which the term used to describe the process changed, refers to the process as CAP.

[12]According to a Region I Branch Chief, the term "single unit" generally refers to a nuclear power plant site with only one operating reactor inside the protected area fence. Although there are two operating units at the Indian Point site (IP2 and IP3), Region I treats IP2 as a single unit site due to its past regulatory performance problems. This results in the allocation of more inspection resources at IP2 than would be the case if the plant were treated as a dual-unit site.

that was occurring during the 1995–1999 time frame. According to NRC Region I staff, between 1995 and 2000, overall IP2 performance was not considered very good. Staff said that during that time period, IP2 had problems related to the plant's CAP.

Region I staff told OIG that it viewed 1995 as a downward turning point for the plant and recalled the grit intrusion event that occurred that year as an example of this decline. Between October 1996 and April 1997, NRC staff conducted four inspections of IP2, which resulted in the issuance of an NOV in May 1997 based on nine violations of NRC requirements. The inspections included an Integrated Performance Assessment Process (IPAP) and three routine inspections conducted by the NRC resident inspectors. Problems identified during the inspections included weaknesses in IP2's design control which, staff explained, pertained to the availability and completeness of design bases information and problems with the CAP.

The Region I Administrator told OIG that following February 1997 there was a series of events that occurred at IP2, coupled with NRC's inspection findings, that reinforced his concerns about IP2's declining performance. He told OIG that the NRC subsequently sent a message to ConEd management by issuing fairly significant civil penalties and a confirmatory action letter (CAL).[13] Additionally, he met with ConEd's Chief Executive Officer to address NRC's concerns about IP2's declining performance, the decline in overall effectiveness of management oversight, and a perception that management tolerated problems rather than aggressively identifying and correcting them.

Consequently, ConEd management responded to Region I by documenting actions it planned to take to arrest the performance declines and to improve the quality of these activities. These detailed action plans were included in a program that ConEd identified as the *Strategic Improvement Program.*

Declining Systematic Assessment of Licensee Performance Scores

According to the Region I Administrator and Region I staff, the Region's concerns about IP2's performance in 1996 were documented in NRC's Systematic Assessment of Licensee Performance (SALP) scores and periodic SALP reports for IP2. The SALP was an NRC evaluation of plant performance conducted every 12 to 24 months within the parameters of NRC's inspection program. The report included a numerical rating of the plant in four categories — plant operations, maintenance, engineering, and plant support — as well as a narrative discussion of performance in each area.

In the SALP report covering the period from September 17, 1995, through February 15, 1997, Region I staff noted that overall performance at IP2 declined. Performance in the areas of operations and plant support were rated as generally effective and some elements were very good; however, performance declined in maintenance and substantively declined in engineering. The SALP report noted many equipment problems were due to the poor condition

[13]CALs are letters issued by NRC to licensees or vendors to emphasize and confirm a licensee's or vendor's agreement to take certain actions in response to specific issues.

of a number of systems. Licensee management was involved in many plant activities and made operational decisions, but management oversight was at times ineffective regarding overall efforts to identify, evaluate, and correct problems.

IP2 Shuts Down To Address Backlog of Problems

OIG learned that following repetitive failures of safety-related electrical breakers, IP2 voluntarily shut down to address a large backlog of equipment, programmatic, and performance problems. This outage lasted from October 1997 until September 1998. According to NRC staff, IP2 used this period to try to better identify and correct these deficient conditions at the plant.

Instead of conducting a planned Operational Safety Team Inspection (OSTI)[14] of IP2, NRC permitted ConEd to hire a team of independent experts to conduct an Independent Safety Assessment (ISA) of the power plant in the spring of 1998. NRC assembled a special NRC Evaluation Team (NET) to gauge the validity and effectiveness of the ISA and review the outcome. The NET observed and evaluated the IP2 ISA from March 30 through May 7, 1998, to assess the validity of the ISA conclusions and to determine whether the ISA had fulfilled NRC's intent to obtain an OSTI-type performance assessment. According to the NET report, the ISA achieved noteworthy insights, including the identification of problems with IP2's CAP. Specifically, the ISA found that the CAP was cumbersome and inefficient, many corrective actions were untimely, and completed actions were typically not revisited to see whether they had achieved their intended impact. According to an NRC staff member who participated in the review, IP2's CAP "was not working very well at all."

Subsequent to the ISA findings, ConEd developed plans to improve station performance and, according to the regional staff and inspection reports, IP2's performance began slowly to improve following plant startup in September 1998. According to the NRC staff, inspection reports, and other docketed correspondence between NRC and ConEd, substantial changes were made to IP2's CAP during this period. However, although progress was made, a number of problems remained that required continued licensee management attention.

IP2 Experiences Two Significant Events

In August 1999, IP2 experienced a reactor trip, or shutdown, a risk-significant event that NRC staff characterized as preventable and partly attributable to weaknesses in IP2's CAP. The reactor trip was caused partly by a condition involving repetitive problems with one channel of RPS's over-temperature/delta-temperature circuitry. The condition, which existed since January 1999, had not been promptly identified, the cause of the condition had not been determined, and corrective actions had not been taken. According to the Region I Administrator, while the August 1999 event challenged safe operation, safety margins were maintained at an acceptable level.

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[14]At the time, OSTIs were conducted to supplement normal inspections for special purposes such as to verify that a plant operator has properly prepared the staff and the plant for resumption of power operations after an extended shutdown. These inspections were performed by either a headquarters or regional team and typically consist of a 2-week onsite inspection conducted by a team of seven inspectors and a team leader.

In February 2000, IP2 experienced yet another significant problem attributed to weaknesses in IP2's CAP: a steam generator tube ruptured in one of its four steam generators, resulting in a leak that allowed pressurized radioactive water, which acts to cool the reactor, to mix with non-radioactive water in the steam generator. The power plant was manually shut down following the event. This resulted in a minor radiological discharge to the atmosphere.

CAP Problems Persist at IP2

In an NRC inspection report (IR 05-247/2001-002) issued in 2001, the Region I inspection team again noted weaknesses in IP2's CAP. According to the report, IP2's progress to effect change continued to be slow. The report "noted problems similar to those that have been previously identified at the IP2 facility, including those in the areas of design control, human and equipment performance, PI&R, and emergency preparedness."

When interviewed by OIG, Region I staff attributed IP2's CAP problems to a large backlog of problems — any one of which might not appear significant. Staff said that IP2 was able to identify problems but was frequently ineffective at prioritizing and correcting them and determining their root cause. Staff attributed this specifically to a cultural problem at IP2 that was not recognized by ConEd management until after the August 1999 event. Staff described this culture as one in which ConEd management did not emphasize or encourage staff efforts to prioritize the correction of problems and identify root causes.

The Region I Administrator and staff acknowledged that the improvements at IP2 were slow, and in some respects limited, but steady. The Region I Administrator told OIG that IP2 met NRC's minimum regulatory requirements and there was never a situation where the margins of safety had been reduced to a point where the plant was unsafe. He added that as a regulator one has to work within the regulatory framework and distinguish between conditions that are unsafe and conditions that involve weaknesses in performance. The Region I Administrator and staff repeatedly emphasized to OIG that the increased inspections and aggressive oversight never identified a situation where IP2 was unsafe.

The Region I Administrator explained to OIG that IP2's rate of improvement above fundamental protection of public health and safety is determined by the plant management. The licensee determines the type and amount of resources that it will apply to facilitate improved performance. The licensee also makes personnel selections at the plant and it is ultimately up to the individuals hired to make these improvements and effect change. The Region I Administrator told OIG that he continually pressed ConEd management to strengthen the margins of safety at IP2 by conducting numerous inspections and special assessments and by communicating the Region's findings to ConEd in a clear and direct manner.

<u>OIG FINDING</u>

Between 1995 and 2001, IP2 experienced a series of operational problems, attributed in large part to deficiencies in IP2's CAP. OIG found that during this period, Region I dedicated significant resources to conduct inspections, document findings, and issue sanctions, yet problems persisted at the plant. Between April 1995 and February 2001, NRC conducted 20 special team inspections at IP2, logging 5,870 inspection hours dedicated to engineering and problem identification and resolution. Furthermore, between 1995 and 2001, Region I issued 13 enforcement actions to IP2. Many of the inspections identified problems with IP2's CAP. However, despite heightened levels of NRC attention to these weaknesses, problems at IP2 remained unresolved. OIG found that in spite of the intensified regulatory oversight by Region I, IP2 was only able to achieve limited improvement in plant performance.

IV. NRC'S UTILIZATION OF THE SMM PROCESS TO HEIGHTEN ATTENTION AT IP2

Senior Management Meeting Process

Between 1986 and 2001, NRC held Senior Management Meetings (SMM) semiannually to allow NRC senior managers to focus agency attention on those plants of highest concern and to monitor licensee efforts to recognize and resolve performance problems. According to the March 1997 version of NRC Management Directive (MD) 8.14, "Senior Management Meeting (SMM)," the primary goal of an SMM was to identify declining trends in the operational safety of individual plants so that early corrective actions could be implemented. OIG was told by senior NRC managers that the SMM offered a means to communicate NRC's concerns to licensees with poor or adverse performance trends.

During the SMM, the senior NRC managers could opt not to take action regarding a particular plant or they could choose to take one of several actions to heighten oversight. For example, senior managers could choose to issue a Trending Letter to advise a licensee that NRC had taken notice of declining plant performance and that if performance did not improve, the plant might be placed on the NRC's Watch List. Or, the managers could choose to place a plant directly on the Watch List. A plant placed on the Watch List received increased oversight from NRC in the form of additional inspections, letters expressing agency concerns about declining performance, and other types of regulatory attention. According to the NRC staff, designation as a Watch List plant could also bring significant public attention to a licensee and could result in a negative economic impact for the utility. These potential negative consequences would motivate a licensee to improve plant performance.

Senior Management Meetings were chaired by the NRC Executive Director for Operations. Participants typically included the Deputy Executive Director for Regulatory Programs; Deputy Executive Director for Regulatory Effectiveness, Program Oversight, Investigations and Enforcement; Deputy Executive Director for Management Services; Regional Administrators; Directors of the Offices of Nuclear Reactor Regulation, Analysis and Evaluation of Operational Data, Nuclear Material Safety and Safeguards, Nuclear Regulatory Research, Enforcement, Investigation, and State Programs; and senior managers from the Office of the General Counsel.

Region I Administrator Seeks SMM Action on IP2

OIG learned that paralleling NRC's inspection activity at IP2 from 1997 through 2000 was a series of attempts by the Region I Administrator to further heighten NRC oversight at the plant through the agency's SMM process. At the June 1997 SMM, the Region I Administrator presented his concerns regarding the declining performance of IP2 that was the result of significant equipment, human performance, and technical support performance issues that were apparent in late 1996. NRC Regional Administrators and senior managers told OIG that at the June 1997 SMM, the Region I Administrator made "a strong presentation" regarding IP2's performance and his belief that IP2 should be designated as a Watch List plant. However, the senior managers decided not to designate IP2 as a Watch List plant but to continue providing the heightened level of regional oversight underway at the time. According to the senior managers, and based on minutes of the SMM proceedings, the information presented at the SMM did not identify a situation where the plant was unsafe, a safety system was inoperable, or adverse trends were apparent. Thus, the senior managers determined that IP2 did not warrant agency-level action.

During the SMM held in January 1998, the Region I Administrator again presented IP2 for discussion asserting that there had been little change in performance in most respects over the prior 6 months; that recent inspections raised additional concerns with respect to performance; that NRC inspectors, rather than ConEd, continued to identify many of the performance problems, particularly in operations and engineering; and that equipment and human performance issues continued to be of concern. Additionally, the informality of processes contributed to problems observed in several areas, including technical specification implementation, procedural adherence, problem identification, and timely effective resolution of issues. OIG learned that this time, the consensus of the senior managers was to conduct a diagnostic-type review to obtain additional information on the plant's condition and not to issue a trending letter or put the plant on the Watch List. Again, the senior managers believed that Region I did not identify a situation where the plant was unsafe or a safety system was inoperable; however, they acknowledged that IP2 continued to exhibit performance weaknesses, and they noted that a definitive improvement trend was not apparent.

In July 1998, the Region I Administrator again presented IP2 at the SMM in the belief it should be designated as a Watch List Plant. He asserted that the performance at IP2 was largely unchanged during the preceding 6 months with respect to human performance and the control of plant activities. Additionally, the 1998 Independent Safety Assessment (ISA) conducted by ConEd identified some important deficiencies and weaknesses that existed at IP2 particularly in the areas of management and operations. Despite the Region I Administrator's presentation, the SMM again declined to designate IP2 a Watch List plant. This time, the SMM decided to maintain, rather than increase, the level of attention to allow the licensee a period of time to execute its performance improvement initiatives. The senior managers recognized that IP2 continued to have performance weaknesses, but again they believed that Region I did not identify a situation where the plant was unsafe or a safety system inoperable.

IP2 was not discussed during the April 1999 SMM. The Region I Administrator told OIG that he did not recommend that IP2 be presented for discussion because it had experienced no significant events since the last time he presented the plant for SMM discussion. He felt that in 1999, performance weaknesses still existed but that IP2 was no worse than in preceding years and was, in fact, slowly improving. He said he still would have preferred SMM action; however, he felt he lacked a basis for presenting the plant at the SMM.

SMM Designates IP2 as Agency Focus Plant in May 2000

In May 2000, the Region I Administrator presented IP2 at the SMM after the occurrence of two significant events at the plant, the August 1999 reactor trip and the February 2000 steam generator tube rupture. OIG learned that overall, the events and related findings during this assessment period represented issues that were of substantial significance; therefore, the senior managers categorized IP2 as an Agency Focus Plant under the revised SMM process.[15]

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[15]In April 1999, the Commission approved SECY 99-086, "Recommendations Regarding the Senior Management Meeting Process and Ongoing Improvements to Existing Licensee Performance Assessment Processes." SECY 99-086 eliminated the "Watch List" and proposed that during SMM meetings, participants would

According to the SMM minutes, the senior managers concluded that the broad performance issues that had existed at IP2 for the past several years revealed a number of deficiencies in the plant's CAP and that IP2 improvement initiatives yielded some progress but, overall, were limited in remedying the underlying problems.

According to the Region I Administrator, the August 1999 and February 2000 events revealed the depth of IP2's performance problems and were evidence of the significant issues discussed at previous SMMs. Region I staff echoed this sentiment to the OIG, questioning why — given the inspection history, the identified problems, the NRC man-hours at the plant, and the history of civil penalties — IP2 was not put on the Watch List sooner.

Current Status of IP2

Region I staff has informed OIG that since March 2001, NRC has provided a significant amount of oversight and inspection effort at IP2. The Region I staff performed 12,950 hours of inspection activity at IP2 between March 1, 2001, and March 1, 2003, compared to an average of 8,297 hours at other single unit sites in Region I. (See Appendix B for a chronology of NRC inspection activity at IP2 during this time period.) Of the 12,950 hours of inspection performed at IP2 during this 2-year period, 2,216 hours were focused on engineering and PI&R compared to an average of 1,077 hours devoted to these areas at other single-unit Region I sites. The staff informed OIG that these figures indicate that during this period, IP2 has received about 1.5 times as much inspection as the average for other single-unit sites and about 2 times as much inspection pertaining to engineering and PI&R.

Annual assessments of plant performance[16] performed since the plant was categorized as an Agency Focus Plant in May 2000 indicate that IP2 performance has been improving, albeit slowly, since that time. NRC's annual assessment of plant performance for April 2, 2000, to March 31, 2001, found that while IP2 met all cornerstone objectives, it remained in the Multiple/Repetitive Degraded Cornerstone column of the NRC's ROP Action Matrix. According to the Region I staff, that assessment noted a number of issues in design control, equipment reliability, PI&R, and human performance. While some performance improvements were noted, progress was considered slow and limited in some areas. Region I staff noted that as of December 31, 2001, IP2 remained in the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix.

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determine whether a plant warranted Agency Focus (characterized by NRC Executive Director for Operations and Commission involvement, e.g., issuance of an order), Regional Focus (managed by the regional administrator, e.g., issuance of a confirmatory action letter), or routine oversight.

[16]Under the ROP, NRC assesses licensee performance in various ways, including quarterly plant performance assessments based on inspection findings and performance indicator data. Regional offices conduct a more comprehensive review after the second quarter of the year (mid-cycle) to assist in planning inspections for the next 6 to 12 months. The regions also conduct an annual (end-of-cycle) review after the fourth quarter of the year to develop an annual performance summary for each plant and to plan inspections for the next 12 months. NRC uses an Action Matrix to assist staff in reaching objective conclusions regarding licensees' safety performance. The matrix allows for plants to be categorized into five possible results categories, or matrix columns, which indicate the plant's level of performance and the agency's required response. Categories (from lowest to highest performance) are (1) Unacceptable Performance, (2) Multiple/Repetitive Degraded Cornerstone, (3) Degraded Cornerstone, (4) Regulatory Response, and (5) Licensee Response.

Significant inspection activity continued during 2002, including an augmented PI&R inspection and supplemental team inspection in June and July 2002. OIG was told that in August 2002, IP2 had made sufficient progress to justify removal of the plant from the Multiple/Repetitive Degraded cornerstone into the Degraded Cornerstone column of the Action Matrix. OIG was told by the Region I Administrator that on February 7, 2003, NRC completed its end-of-cycle plant performance assessment of IP2 covering performance from January 1, 2002, through December 28, 2002. NRC concluded that during that time period, IP2 continued to operate in a manner that preserved public health and safety.

The Region I Administrator and staff told OIG that Region I fully utilized the regulatory tools it had available to deal with IP2. The Region I Administrator said that although the plant was never unsafe, improvement in IP2's performance might have been swifter had the plant been designated a "Watch List" plant by the SMM earlier. This designation would have sent a powerful message to the licensee concerning the need for improved performance.

The Region I Administrator commented that while the agency's senior managers designated the plant as an "Agency Focus Plant" in May 2000, this occurred after the plant had reversed its downward trend and, in fact, the designation had a relatively small impact on recent plant operations because the plant's declining performance had already been arrested as a result of earlier actions taken by the NRC. The Region I Administrator also noted that SMM deliberations were always thorough but that decisions were inherently difficult given the complexity of issues involved.

Additionally, the Region I Administrator commented to OIG that Entergy's purchase of IP2 in September 2001, had a considerable impact on plant performance. According to the Region I Administrator, Entergy conducted its own self-assessment of IP2 and subsequently committed significant resources to the plant. Furthermore, Entergy had experience operating other nuclear power plants, was aware of the need to inject resources to improve plant performance, and had those resources available. Entergy also understood the need to bring top management talent to operate the plant, which it did. According to the Region I Administrator, this shift in ownership facilitated the IP2 improved performance trend.

The Region I Administrator considered IP2's improvement as an NRC "regulatory success story." He stated that NRC's aggressive oversight and intervention arrested the decline in early 1996 and prevented IP2 from ever getting to the point where it was unsafe to operate. He acknowledged that IP2's improvement has been slow at times and often uneven, but that, overall, plant performance has steadily improved. In his view, the conditions that led to IP2's poor performance in the mid-1990s developed over a number of years and, therefore, required time to resolve. He credited NRC oversight efforts performed at IP2 since 1996 with having caused the plant to reverse its downward performance trend and begin its slow progress toward the performance improvement reflected in the NRC's recent assessment letters.

OIG FINDING

On four occasions between 1997 and 2000, the Region I Administrator sought additional NRC oversight for IP2 by seeking to have NRC's senior managers place IP2 on NRC's Watch List via the agency's Senior Management Meeting process. However, it was not until May 2000, after the August 1999 reactor trip and the February 2000 steam generator tube rupture, that NRC senior managers agreed that this form of heightened attention was appropriate. In May 2000, IP2 was classified as an Agency Focus Plant. Subsequent to being so designated, NRC annual assessments of plant performance indicated that IP2 had improved. OIG concurs with the Region I Administrator and his staff that placing IP2 on the Watch List sooner might have sufficiently motivated the licensee to cause earlier improved performance.

APPENDIX A
Summary of IP2 RPS Condition Reports

CR 199803574 identified a discrepancy between the RPS wiring configuration and a description in section 7.2.2.9 of the UFSAR of isolation between safety signals and annunciator and/or computer signals. Contrary to the UFSAR statement that "The center and front decks of RPS logic relays are used for annunciator and computer signals respectively," 22 RPS logic relays were found to violate this criterion.

CR 199900478 identified discrepancies between design drawings and the as-built configuration with respect to contact state associated with interposing relays for the low autostop oil pressure protection scheme. The corrective action for this condition involved revision of four drawings to reflect the field condition.

CR 199902274 identified "minor" inconsistencies affecting 14 RPS and ESF drawings. Corrective action involved revising the affected drawing based on comments received from an outside contractor who was tasked with the drawing review.

CR 199902835 identified three distinct discrepancies between plant drawings and the as-built condition. These discrepancies involved: RPS logic relays used to block the "Source Range High Influx at Shutdown" annunciator, drawings showing RPS relay contact configuration different from the as-built condition, and incorrect RPS relay nomenclature on plant drawings. The corrective action for this CR was limited to revising the affected drawings to agree with the as-found condition.

CR 199903445 was initiated because the drawing revisions prepared in response to CR 199902835 were in error. This CR also identified an additional drawing error in which the drawing showed the incorrect RPS relay contacts used for the Source Range High Flux at Shutdown annunciator block.

CR 199904968 identified another discrepancy between the design drawings and the as-found configuration of the RPS. This discrepancy involved contacts from RPS relay P10-2 that are used to defeat the Source Range Loss of Detector Voltage annunciator above 10% reactor power which are not shown on plant drawings. The corrective action for this CR involved a field verification of the configuration and revision of the affected drawing to reflect the as-found condition.

CR 200007597 identified a number of potential internal wiring related discrepant conditions in the reactor protection racks. Isolated cases of wire routing and/or terminations were observed to be inconsistent with routing/separation requirements stated in the UFSAR. In response to this CR an Operability Determination (OD) 00-018 was issued to address the wiring routing/separation issues. The OD determined that the RPS was operable.

CR 200008415 identified drawing discrepancies between Westinghouse RPS wire lists and field conditions, however, an operability determination concluded that this did not constitute an operability concern.

CR 200008818 identified a broken contact in a reactor trip relay, unidentified, unterminated switchboard wire with exposed lugs in RPS cabinets, and a mixing of wiring associated with computer/logic/annunciator functions. The broken contact has been repaired. A 200-degree hold was placed on this CR. The "Operability Review Note" by the Watch Engineer stated "200 degree hold for loose wires, etc." The response to the unterminated (loose) wire issue was not addressed. The engineer who responded to the 200H action stated that he considered the unterminated wire a housekeeping issue and therefore, did not address it as part of the 200H response.

CR 200009499 identified additional conditions in which the wiring in the RPS racks violated statements in the UFSAR. The CR stated that "Wires (in RPS Racks 4 and 5) were carelessly strewn through multiple wire ways," and "Had the original design been followed, there would have been no mixing (of circuit functions) and there would have also been half as many new wires to mix." These issues were addressed in Operability Determination 00-018 which was conducted on CR 200007597 which found that the RPS was operable.

CR 200009641 identified six issues related to RPS wiring deficiencies or discrepancies, three of which were similar to or a repeat of issues identified in previous CRs. The new issues included a wire associated with an NIS power range logic relay with a splice that is not represented on plant drawings and single cable containing both 125 VDC logic protection power and 118 VAC instrument bus power. Both of these issues were addressed in Operability Determination 00-018.

CR 200010125 identified discrepancies between design drawings and the as-built configuration of the RPS. This CR also identified other CRs that described similar inconsistencies between design drawings and RPS wiring. A review of the corrective action associated with these CRs revealed that the CR actions were typically closed by revising the plant drawings to reflect the as-found configuration without performing a safety evaluation to determine the impact of the change on the design and licensing basis. In some cases the as-found condition affected the system design as depicted in the UFSAR text and/or figures. This CR also identified errors made in drawings as part of the corrective action for CR 199904968. Furthermore, this CR identified discrepancies between drawings and the as-found RPS wiring that had not been previously identified.

CR 200100327 summarized numerous issues identified in eight previously submitted CRs that documented a lack of configuration control and quality control of changes to the RPS wiring since 1998. The concerns raised in CR 200100327 were categorized as quality assurance requirements for design verifications, wiring changes resulting from modifications that could not be located and wiring separation not in accordance with the UFSAR. The eight CRs summarized in CR 200100327 are CR 200010125, CR 199803574, CR 199904968, CR 199902835, CR 199903445, CR 200007597, CR 200009499 and CR 200009641.

APPENDIX B
Chronology of Significant Inspections and Oversight at IP2, 1995 – 2003[1]

March 14, 1995	Inspection Report (IR) 1995-01, special safety inspection of AFW digital controller failure.
April 12, 1995	IR 1994-017 service water self-assessment inspection.
August 28, 1995	IR 1995-080, Operational Safety Team Inspection.
October 26, 1995	SALP report issued.
January 28, 1997	IR 1996-080, Integrated Performance Assessment Process (IPAP).
January 31, 1997	Confirmatory Action Letter (CAL) issued.
February 21, 1997	CAL closed.
March 31, 1997	Final SALP report issued.
May 1, 1997	Plant shutdown for refueling outage.
May 9, 1997	IR 1997-003 integrated inspection.
June 19, 1997	IR 1997-005, special inspection conducted for stuck open MSSV.
June 1997	IP2 discussed at Senior Management Meeting (SMM).
June 1997	Regional Administrator meets with ConEd Chief Executive Officer.
July 8, 1997	Plant startup from refuel outage.
July 26, 1997	Generator load rejection and reactor trip.
July 28, 1997	Reactor trip.
August 6, 1997	Shutdown.
August 8, 1997	IR 1997-008, special inspection of outage issues.
August 23, 1997	Reactor trip due to reactor coolant pump breaker testing logic error.
August 25, 1997	Plant startup.
September 29, 1997	IR 1997-010, special inspection of load reject and reactor trip.

[1]Information in this chronology was provided to OIG by Region I.

October 14, 1997	Plant shut down due to repetitive DB50 circuit breaker failures.
December 12, 1997	IR 1997-012, integrated inspection report, resident inspection and specialist review of safety-related breaker problems.
January 1998	IP2 discussed at SMM.
January 1998	Performance letter issued to ConEd - decision made to perform Operational Safety Team Inspection/Independent Safety Assessment.
February 13, 1998	IR 1997-013, special inspection of 480 Vac Breaker failures.
March 26, 1998	CAL 1-98-005 due to issues discovered during shut down not related to circuit breakers.
March 26, 1998	IR 1998-201, design inspection.
April 27, 1998	NRC restart action plan for IP2 issued.
May 1998	Independent Safety Assessment performed by ConEd.
June 3, 1998	IR 1998-005, NRC Evaluation Team (NET).
June 26, 1998	IR 1998-006, special inspection focusing on corrective actions regarding plant restart issues.
June 1998	Emergency preparedness exercise.
July 9, 1998	Revised NRC CAL 1-98-005 issued March 26, 1998.
July 1998	IP2 discussed at SMM.
September 16, 1998	IR 1998-012, followup NRC NET evaluation team inspection.
September 21, 1998	Reactor startup.
October 16, 1998	IR 1998-008, special Inspection of corrective action associated with restart issue.
October 23, 1998	IR 1998-014, NRC integrated inspection.
November 3, 1998	IR 1998-016, NRC special inspection of high efficiency particulate air (HEPA) filter deterioration.
January 29, 1999	IR 1998-018 NRC 40500 Corrective Action Program Inspection.
April 1999	Plant Performance Review.

June 1999	ConEd external assessments of operations, work control, and maintenance departments.
August 19, 1999	IR 1999-004 NRC team inspection report (Core Engineering Team).
August 31, 1999	Reactor trip and loss of offsite power.
September 14, 1999	Management meeting - Augmented Inspection Team (AIT) interim results.
September 23, 1999	Public exit meeting - AIT exit meeting.
September 1999	Emergency preparedness exercise.
October 13, 1999	Reactor startup.
October 19, 1999	IR 1999-008, AIT.
October 1999	IR 1999-013, AIT follow up team inspection commenced.
October 1999	Mid-cycle plant performance review letter issued.
November 23, 1999	Public Meeting - IP2 performance assessment results from September 1999 plant performance review.
December 21, 1999	Results of the follow-up inspection to the AIT (1999-013).
December 1999	IP2 Recovery Plan actions transferred to Business Plan.
January 5, 2000	IR 1999-014. Results of enforcement follow up of AIT for August 31, 1999 trip.
January 7, 2000	Drafted charter for the formation of the Indian Point Unit 2 oversight panel (IPOP).
February 1, 2002	Drafted IP2 oversight strategy.
February 15, 2000	Reactor trip - steam generator tube failure (SGTF).
March 1, 2000	SGTF meeting.
March 14, 2000	SGTF public meeting.
March 2000	Formation of IP2 communications team.
March 2000	Plant performance review letter.
April 28, 2000	NRC AIT SGTF IR 2000-002 issued.

May 23, 2000	IP2 discussed at SMM; letter issued characterizing IP2 as an "Agency Focus" plant.
June 25, 2000	Public meeting.
July 10, 2000	IR 2000-007, AIT SGTF follow-up.
July 27, 2000	IR 2000-010, NRC SGTF special inspection.
August 3-4, 2000	Regional Administrator site visit.
August 31, 2000	IR 2000-010, SGTF special inspection.
September 11, 2000	NRC Agency Focus Meeting. (Regional Administrator and NRR Deputy Director Site Visit)
September 26, 2000	Regulatory conference on SGTF "red" finding.
September 2000	Ongoing regional management briefings on cornerstone deficiencies, and plant performance issues throughout restart.
October 2, 16, 2000	Problem Identification and Resolution (PI&R) inspection.
October 5, 2000	EDO brief to discuss content of "Agency Focus" letter.
October 10, 2000	Assessment follow up (Agency Focus Update) letter.
October 11, 2000	ROP meeting held in Cortland Town Hall.
October 16, 2000	Operator requalification Inspection.
October 25, 2000	NRC - ConEd management meeting.
October 31, 2000	Significant Determination Process repanel (final determination of "red or yellow" finding for SGTF issues).
November 1, 2000	IP2 SGTF Lessons Learned Task Force (LLTF) report issued.
November 6, 2000	NRC on-site restart readiness reviews.
November 8, 2000	Mid Cycle review meeting conducted.
November 14, 2000	RI review of four system readiness reviews.
November 16, 2000	Public meeting.
November 16, 2000	NRC noted that the independent 125 VDC SSFA team performed a high quality review.

November 20, 2000	Issued red finding and Notice of Violation (NOV) for the poor SG inspection program that led to the SGTF.
November 27, 2000	NRC safety system readiness review inspection on the Safety Inspection system.
November 29, 2000	Mid cycle performance review and inspection plan letter issued.
December 1, 2000	Region I senior management site visit to IP2.
December 4, 2000	PI&R inspection report.
December 6, 2000	EDO briefing.
December 11, 2000	Plant heat up above 200 degrees - restart inspection begun.
December 18, 2000	IR 2000-014 design issues inspection.
December 20, 2000	NRC replied to ConEd's request for extension to respond to the red finding and NOV.
December 22, 2000	NRC Region I issues NRC review efforts/status letter.
December 30, 2000	Plant restarted.
January 2, 2001	Turbine trip due to low SG level.
January 5, 2001	Regional Administrator visits Congresswoman Kelly.
February 9, 2001	95003 multiple degraded cornerstone supplemental inspection.
February 26 - May 4, 2001	IR 2001-005, review reactor protection system (RPS) design issues.
February 27, 2001	Chilling effect letter issues.
March 1-2, 2001	Regional Administrator site visit and public exit meeting for 95003 inspection.
March 9, 2001	Chairman site visit with Regional Administrator and Executive Director for Operations.
April 3, 2001	Division of Reactor Safety (DRS) branch chief visit to IP2 - UFSAR verification project status.
April 10, 2001	IR 2001-002, (95003 Inspection) supplemental inspection report issued.

June 18, 2001	IR 2001-007, emergency preparedness (EP) exercise review and supplemental inspection of licensee actions to address three findings in the EP cornerstone area.
July 23, 2001	IR 2001-007, review of 2001 design engineering business plan and scope and 50.54 (f) commitment status.
July 23, 2001	IR 2001008, review of 2001 Design Engineering Business Plan Scope and 50.54(f) commitment status.
October 22, 2001	IR 2002013, NRC on-site to do initial inspection of the failure of three of six crews on licensed operator (LOR) examinations and to observe facility evaluate seventh crew; crew fails: four of seven = yellow finding.
November 5, 2001	IR 2001-010, review of licensee's safety injection (SI) safety system functional assessment (SSFA) and PI&R inspection.
November 27, 2001	IR 2001-011, NRC observes facility-led evaluation of an operating crew; while onsite, conducts regular-hours control room (CR) observations.
December 7, 2001	IR 2001-011, NRC- led evaluation of another operating crew; while onsite, conducts regular-hours CR observations.
December 16, 2001	IR 2001-011, NRC- led evaluation of 4 staff RO licenses.
January 28, 2002	IR 2001-014, review of licensee's self assessment and Fundamentals Improvement Plan (FIP), including the Design Basis Initiative (DBI).
February 7, 2002	IR 2002-007, NRC observes facility-administered evaluations (High Intensity Training (HIT).
March 21, 2002	IR 2002-007, NRC observes facility-administered evaluations (HIT).
March 21, 2002	IR 2002-009, supplemental inspection to review causes and corrective actions for yellow finding related to operator requalification.
June 24, 2002	IR 2002-010, augmented PI&R inspection, reviewed performance issues related to the multiple degraded cornerstone designation, progress implementing the FIP, and review of the degraded control room west wall fire barrier.
November 4, 2002	IR 2002-007, review of reactor protection system (RPS) wiring verification.
December 9, 2002	IR 2003-002, PI&R team inspection.
December 2002 - February 2003	IR 2003-003 and IR 2003-005 (both draft), team inspections to review TI 2515/148 and various other security issues.

January 27, 2003	IR 2003-004 (draft), engineering team inspection reviewed design and performance capability of component cooling water and offsite power supplies.

APPENDIX C
Summary of Escalated Enforcement Action from 1995-2000

1996-01, Enforcement Action 96-089, Significance Level (SL) III
10 CFR 50.59 (SL III) and 50.72 (SL IV)

Repair activities on central control room roof left ventilation system in unanalyzed condition for 2 months. Inadequate corrective actions.

1996-04, Enforcement Action 96-272, SL IV
Criterion XVI (SL IV) and Technical Specification (TS) 6.8.1. (SL IV)

1) Failure to maintain proper configuration control over containment isolation valve, contrary to procedure requirements.
2) Failure to preform required safety evaluation on procedure change.

1996-07, Enforcement Action 97-031, SL III ($50,000 civil penalty)
Criterion XVI (SL III)

Inadequate measures were taken to assure that the cause of each condition was determined and corrective action taken to preclude repetition.

1) Repeated surveillance test failures associated with the TDAFW pump's steam admission valve and discharge flow control valves. Valve damage subsequently identified.
2) Preconditioning of TDAFW pump by blowing down steam traps prior to testing. Adequate engineering review was not performed to support pump operability.
3) Multiple surveillance test failures associated with alternate safe shutdown system power transfer switches for the 23 and 24 service water pumps.
4) Untimely identification of degradation of PAB filter/fire deluge system control panel and associated circuits. System was incapable of performing design function. Poor implementation of an alarm response procedure's required actions.

1996-08, Enforcement Action 97-113, SL III ($50,000 civil penalty)
Criterion XVI (SL III), TS 6.8.1(SL IV), TS 6.5.1.6.a. (SL IV)

1) Failure to take adequate corrective actions following grit intrusion during the 1995 refueling outage. Resulted in inoperability of three of the four safety-related MFRV's and one low-flow bypass MFRV in January 1997.
2) Control of SG levels not in accordance with procedure and the failure to make temporary procedure changes to invoke administrative allowances for situation where deviation is necessary.
3) Failure to perform a required review of a vendor report that was used as the basis to support DG operability following the 1995 grit intrusion.

1996-80, Enforcement Action 96-509, SL III ($50,000 civil penalty)
Appendix R (SL III)

Fire protection features not provided to protect one train of systems - two instances.

1) Certain normal safe shut down instrumentation and the corresponding alternate safe shutdown instrumentation would be subject to fire damage.
2) Potential for hot shorts exists as a result of fire damage to cables associated with both the pressurizer PORV and block valves (a high/low pressure interface).

1997-03, Enforcement Action 97-191, SL III ($55,000 civil penalty)
Criterion XVI (SL III)

Failure to promptly identify and take corrective actions. Maintenance worker drilled into an electrical junction box, causing fire dampers in two safety-related electrical distribution rooms to actuate. Some dampers did not drop and other became physically restrained and only partially dropped. Condition went unaddressed by plant personnel for two days until questioned by NRC.

1997-08, Enforcement Action 97-367, SL III ($110,000 civil penalty)
TS 6.8.1 (SL III), Criterion XVI (SL III), TS 3.1.A.4.a (SL III), TS 4.18.c (SL III), TS 4.2.1.(SL IV) - 5 violations

1) operation of the plant for 2.5 days outside technical specifications pressure and temperature curves with the OPS inoperable. Violation of TS 6.8.1.
2) Failed to consider ambient temperature condition on the pressurizer code safety valve set point. Violation of TS 4.2.1 Untimely and ineffective corrective actions. Inadequate 50.59 safety evaluation for a plant mod to remove the pressurizer block house roof. Inoperability of the code safety valves as prescribed by the technical specifications. Numerous opportunities existed for the staff to identify this issue.
3) Ingestion of hose in 21 recirculation pump. Poor engineering resolution to degraded pump performance that preceded the identification of the hose in the 1997 refueling outage. Indications are 21 recirculation pump inoperable since 1995. Inadequate corrective actions.

1997-13, Enforcement Action 97-576, SL III ($55,000 civil penalty)
Criterion XVI (SL III)

Failure to take prompt and appropriate corrective actions prior to voluntary shutdown in October 1997 to address the recurring DB-50 breaker failures to close on demand.

1997-15, Enforcement Action 98-028, SL IV
Criterion XVI (SL IV), TS 6.8.1 (SL IV) - 2 violations

1) ConEd's failure to address degraded conditions in a timely manner on the post accident containment venting system (PACVS) and the hydrogen recombiner system.
2) An inadequate procedure for operation of the PACVS.

Office of Investigations- January 22, 1998, Enforcement Action 98–056, SL III
50.9 (SL III) - 2 Violations

1) On August 8, 1997, the emergency battery lights in the PAB were not tested per procedure. However, records were created that indicated the lights were tested. Technicians were not in room for long enough period to adequately test lights.

2) On August 8, 1997, surveillance test of EDG auxiliaries require double verification. Double verification of compressor was not performed. Records were created that indicate second verification was performed. Technician was not in the EDG building to be able to perform verification.

1998-02, Enforcement Action 98-192, SL III ($55,000 civil penalty)
Criterion XI (SL III)

A significant number of technical surveillance testing discrepancies were identified through ConEd and NRC reviews. Failed to assure that all testing required to demonstrate that systems and components will perform satisfactorily in service, as specified in technical specifications, was incorporated into surveillance test procedures.

1999-014, Enforcement Action 99-319, SL II ($88,000 civil penalty)
Criterion III (2 violations), Criterion V, Criterion XVI (SL II)

1) a. Design basis not correctly translated into specifications and procedures for mod to the 480 vital bus degraded voltage relays. Therefore, relays could not perform design basis function and correctly reset. Contributing to August 31, 1999 transfer of 480V bus from offsite power supply to the RDGs.
b. Requirement for auto operation of the Station Aux Transformer Load Tap Changer were not translated into procedures. As a result form September 9, 1998 to August 31, 1999, the 138kV offsite power system was unable to perform its function. Violated Technical specification 3.7. B.3.
2) Procedure did not adequately ensure proper calibration of DB-75 breaker trip units for the EDGs. Result EDG was inoperable from May 27, 1999 through August 31, 1999.
3) Condition adverse to quality with channel 4 of the reactor protection system (RPS) OTDT circuitry between January 1999 and August 31, 1999, resulting in a plant trip during maintenance on channel 3.

2001-010, Enforcement Action 00-179, Red Finding
Criterion XVI (Red)

A PWSCC defect was identified, signifying the potential for other similar cracks in low-row tubes. ConEd did not adequately evaluate the susceptibility for low-row tubes to PWSCC and the extent of degradation.
ConEd did not adequately evaluate the potential for hour-glassing based on the indications of the low-row tube denting. The increased stresses caused by the hour-glassing are a prime precursor for PWSCC.
1997 Steam generator inspection program was not adjusted to compensate for the

adverse effects of increased noise in detecting flaws, particularly when condition that increased the susceptibility to PWSCC existed.

- These problems contributed to at least four tubes with PWSCC flaws in their small radius U-bends, being left in service following the 1997 inspection, until one tube failed on February 15, 2000.

EXHIBIT X

April 25, 2003

MEMORANDUM TO: Chairman Diaz

FROM: Hubert T. Bell
Inspector General /RA/

SUBJECT: NRC ENFORCEMENT OF REGULATORY REQUIREMENTS AND COMMITMENTS AT INDIAN POINT, UNIT 2 (CASE NO. 01-01S)

Attached is an Office of the Inspector General (OIG), U.S. Nuclear Regulatory Commission (NRC) Event Inquiry that addresses the NRC's oversight of operations at the Indian Point, Unit 2 nuclear power plant in Buchanan, New York.

Please call me if you have any questions regarding this Event Inquiry. This report is furnished for whatever action you deem appropriate. Please notify this office within 90 days of what action, if any, you take based on the results of the Event Inquiry.

Attachment: As stated

cc w/attachment:
Commissioner Dicus
Commissioner McGaffigan
Commissioner Merrifield
W. Travers, EDO

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04/ **25** /03	04/ **24** /03	04/ **25** /03	04/ **25** /03	04/ **25** /03

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OFFICE OF THE INSPECTOR GENERAL
EVENT INQUIRY

UNITED STATES NUCLEAR REGULATORY COMMISSION

NRC ENFORCEMENT OF REGULATORY REQUIREMENTS AND COMMITMENTS AT INDIAN POINT, UNIT 2

Case No. 01-01S

/RA/
Veronica O. Bucci, Special Agent

/RA/
George A. Mulley, Jr.,
Senior Level Assistant for
Investigative Operations

/RA/
Brian C. Dwyer
Assistant Inspector
General for Investigations

NRC ENFORCEMENT OF REGULATORY
REQUIREMENTS AND COMMITMENTS
AT INDIAN POINT, UNIT 2

Case No. 01-01S   April 25, 2003

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BASIS AND SCOPE

The Office of the Inspector General (OIG) initiated this inquiry in response to a Congressional request that OIG examine issues concerning U.S. Nuclear Regulatory Commission (NRC) oversight of operations at the Indian Point 2 (IP2) nuclear power facility in Buchanan, New York. The request referred specifically to "internal Con Ed/Indian Point 2 condition reports" made public in a January 2001 petition review board meeting that "may include information which indicates that the plant operator may be in violation of a commitment made back in 1997 regarding design bases requirements."

The Congressional request also focused on issues raised by an engineering consultant hired by the licensee who had recently resigned his position due to a differing professional opinion regarding the plant's Reactor Protection System. The request noted that one of the more lengthy condition reports cited discrepancies between design drawings and the as-built configuration of the Reactor Protection System.

Based on the above concerns, OIG initiated an Event Inquiry to examine:

I. NRC's oversight of IP2's progress toward fulfilling two design bases commitments made to the NRC in 1997. These commitments were made in response to NRC's 1996 request for information concerning plant programs and processes for controlling and maintaining operations within the facility's design bases.

II. NRC's response to the specific concerns raised by an IP2 engineering consultant pertaining to discrepancies between design drawings and the as-built configuration of the Reactor Protection System.

III. NRC's oversight of IP2's corrective action program between 1995 and 2001.

IV. NRC's utilization of its Senior Management Meeting process to heighten attention to IP2.

BACKGROUND

NRC's Regulation of Power Plants — Overview of Terms Used in This Report

Nuclear power plants are required to adhere to U.S. Nuclear Regulatory Commission (NRC) regulations to ensure their safe operation. These regulations include requirements that power plants operate in accordance with their current license, which includes (1) the plant's technical specifications, (2) license conditions, (3) licensee commitments made in response to NRC Generic Letters and Bulletins, and (4) the Final Safety Analysis Report (FSAR).[1] Design bases information identifies the specific functions to be performed by a power plant's structures, systems, and components as well as associated design parameters.

In addition, plants are required to have a corrective action program (CAP) that enables them to identify, prioritize, and correct problems in a timely manner. Power plants manage their CAP by maintaining a database of action items, or condition reports, which describe particular plant conditions in need of repair or attention. Plants typically prioritize these condition reports based on safety significance and address them accordingly.

NRC provides oversight of nuclear power plants to ensure that plants are operating safely. The agency conducts reactor inspections to determine whether power plants are in compliance with agency requirements. Inspections range from routine, baseline inspections[2] to inspections beyond the baseline which may focus on areas of declining plant performance. The agency issues sanctions (i.e., enforcement actions) — such as Notices of Violation (NOV),[3] fines, or orders to modify, suspend, or revoke licenses — when plants are out of compliance. In 2000, NRC implemented a Reactor Oversight Process (ROP), which was intended to be substantially different from the previous oversight process and to take into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and evaluating the safety performance of NRC-licensed plants. Under this process, inspection findings are evaluated for risk significance using pre-established criteria. Plants that fail to meet certain safety objectives, as determined by performance indicators and inspection findings, are to receive increased inspection activity, focusing on areas of declining performance and may be subject to enforcement action.

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[1]The FSAR is a licensing document that provides a description and safety analysis of the site, the design, design bases and operational limits, normal and emergency operation, potential accidents, predicted consequences of such accidents, and the means proposed to prevent or mitigate the consequences of such accidents. When the FSAR has been updated, it is referred to as the updated FSAR, or UFSAR.

[2]Baseline inspections are common to all nuclear power plants; NRC's baseline inspection program is the normal inspection program performed at all nuclear power plants. The program focuses on plant activities that are "risk significant," that is, those activities and systems that have a potential to trigger an accident, can mitigate the effects of an accident, or increase the consequences of a possible accident.

[3]An NOV formalizes a violation by identifying a requirement and how it was violated.

Between 1986 and 2001, NRC also used its semiannual Senior Managers Meetings (SMM)[4] as a means to increase attention to plants with persistent operational problems. During these meetings, the agency's senior managers reviewed certain plants experiencing declines in performance. Participants decided whether to increase oversight of subject plants and, if so, by what means. For example, a SMM decision might require a plant to undergo additional inspections, or the staff could issue a "trending letter" to advise a licensee that NRC had taken note of declining plant performance, or designate the plant as in need of heightened NRC attention (e.g., designation as an Agency Focus Plant).

One way in which nuclear power plants fulfill NRC expectations is through regulatory commitments. Regulatory commitments are non-binding statements made by licensees to NRC indicating they will take specific actions, for example, to verify the accuracy of UFSAR information, and they typically reflect the means by which licensees will accomplish the commitment (e.g., in a certain timeframe, following a specific approach).

The Indian Point Nuclear Power Plant, Unit 2 (IP2), is one of two operating pressurized water reactors located in Buchanan, NY, 24 miles north of New York City. IP2 began commercial operations in August 1974. The Consolidated Edison Company of New York, Inc. (ConEd), owned IP2 until September 6, 2001, when the plant was purchased by Entergy Nuclear Operations, Inc. NRC's Region I office[5] provides oversight for IP2.

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[4]The Senior Management Meeting (SMM) program which required semiannual meetings of NRC senior managers was replaced in 2001 by the Agency Action Review Meeting (AARM) program. The AARM is an annual meeting of NRC senior managers under the Reactor Oversight Process. This meeting essentially replaces the SMM under NRC's previous oversight process.

[5]NRC has four regional offices that conduct inspections of nuclear reactors within regional boundaries. NRC's Region I provides regulatory oversight for IP2 and other nuclear facilities within the northeast region of the United States.

DETAILS

I. NRC OVERSIGHT OF IP2'S PROGRESS TOWARD FULFILLING TWO 1997 DESIGN BASES COMMITMENTS

Overview of Design Bases

Nuclear power plants are designed so that internal and external events (e.g., loss of coolant accident, fire, earthquake) will not jeopardize plant safety or threaten the health and safety of the public. A plant's design bases in part describe how the plant will cope with various accidents and emergencies. Plant structures, systems, and components (SSC) must be built in accordance with design requirements that will enable the plant to meet its design bases and, consequently, to withstand such accidents and emergencies. Plant operators are expected to not make plant modifications to safety related systems without having performed NRC required safety analyses, which are needed to prove the modification will not affect the plant's ability to meet its design bases requirements. Furthermore, when modifications are made, they are supposed to be reflected in the plant's design bases documents, which link each plant SSC to its design bases and original design requirements. Design bases documents include such information as industry, regulatory, and manufacturer criteria for plant systems and information generally contained in the UFSAR specifying system functions and requirements, component functions and requirements, interface requirements from supporting and supported systems, applicable accident analysis assumptions related to the systems, and plant design drawings and calculations.

NRC Requests Licensee Feedback on Design Bases Issues

NRC team inspections during 1995 and 1996 identified concerns regarding the ability of NRC licensees to maintain and implement the design bases at certain plants. To learn more about the scope and extent of the problems among operating nuclear power reactors, the staff proposed that all licensees be required to provide information regarding the availability and adequacy of design bases information. To that end, on October 9, 1996, NRC issued a letter to each NRC reactor licensee in accordance with Title 10, Part 50, Section 54(f), Code of Federal Regulations (10 CFR 50.54(f)) requesting that each licensee submit under oath a written response within 120 days describing and discussing the effectiveness of its programs and processes for controlling and maintaining operations within the facility's design bases. The stated purpose of the letter was "to require information that will provide the U.S. Nuclear Regulatory Commission (NRC) added confidence and assurance that [licensee plants] are operated and maintained within the design bases and any deviations are reconciled in a timely manner."

Specifically, NRC found it problematic that some licensees had failed to (1) appropriately maintain or adhere to plant design bases, (2) appropriately maintain or adhere to the plant licensing basis, (3) comply with the terms and conditions of licenses and NRC regulations, and (4) assure that the UFSARs properly reflect the facilities. According to the letter, "The extent of the licensees' failures to maintain control and to identify and correct the failures in a timely manner is of concern because of the potential impact on public health and safety should safety systems not respond to challenges from off-normal and accident conditions."

NRC Reviews Overall Response

Subsequent to NRC's receipt and review of all licensee responses to the October 9, 1996, letter, the staff issued SECY-97-160,[6] which described a four-phased approach which NRC had undertaken to review the licensee responses to the 10 CFR 50.54(f) request. The SECY described the completion of the first three phases and concluded that all licensees had established programs and procedures to maintain the design bases of their facilities. However, SECY-97-160 also recommended certain plant-specific, final-phase followup activities to address the staff's concerns about either (1) the performance of certain licensees in controlling facility design bases or (2) the need to validate the effectiveness of a particular element of a licensee's design control program.

A manager in the NRC's Office of Nuclear Reactor Regulation (NRR) told OIG that the request began with a high level of agency concern that there were widespread problems pertaining to the accuracy of plant UFSARs and there was a heightened awareness that these problems needed to be resolved as quickly as possible. However, as licensee efforts to address these concerns unfolded, NRC staff recognized that this effort was more resource intensive than had initially been anticipated, and staff allowed licensees to have more time to complete these efforts.

IP2 Responds to NRC Design Bases Request

In response to NRC's October 1996 10 CFR 50.54(f) request to ConEd regarding IP2, the licensee made two specific commitments. In its February 13, 1997, letter that conveyed these commitments to NRC, ConEd stated its intent "to voluntarily initiate and complete" an UFSAR review program. The program was scheduled for completion within 24 months. The UFSAR review program was to include (1) verification of the accuracy of the UFSAR design bases information, (2) assessment to confirm that the UFSAR design bases information was properly reflected in plant operation, maintenance, and test procedures, (3) review of the UFSAR to identify and resolve any internal disagreements or inconsistencies which could impact the design bases, and (4) development of a process to enhance overall the UFSAR accessibility. In its second commitment, ConEd stated it would continue its "Design Basis Document (DBD) Initiative" to review and update existing design bases documents and create new ones if needed. The continuation of the DBD Initiative was to include supplementation of 22 DBDs with a combination of additional DBDs and added information on interfacing systems. This effort was also to be completed in 24 months.

IP2 Extends Completion Date

In a letter dated February 17, 1999 (24 months after the initial commitments were made), ConEd provided an update to NRC concerning the commitments it had made pursuant to NRC's 1996 request. The letter reported that both the UFSAR verification effort and DBD initiative were underway; the UFSAR effort was approximately 65 percent complete and the

[6]SECY 97-160, "Staff Review of Licensee Responses to the 10 CFR 50.54(f) Request Regarding the Adequacy and Availability of Design Bases Information," dated July 24, 1997.

supplementation of 6 of 27 DBDs was in progress. The letter also changed the completion date of both commitments: December 31, 1999, for the former and December 31, 2002, for the latter.

OIG learned that NRC is not expected to formally approve changes in commitment completion dates such as the one described above. According to the NRR manager, commitments are often schedule or process related (e.g., licensee commitment to fix something by a specific time or in a particular manner) and changes in completion dates are not necessarily problematic. For example, the manager said, a rule may say to fix something in a timely manner and the licensee will commit to do so within 2 months. However, if the licensee fails to make the 2-month deadline, the licensee may adjust the timeframe to another date that NRC would consider timely.

The NRR manager and Region I staff told OIG that after IP2 became involved in these efforts, all parties realized that the 2-year timeframe that ConEd initially committed to was unrealistic. A number of plants, including IP2, required additional time to complete their review and NRC staff generally viewed these extensions as reasonable.

OIG also learned that with regard to ConEd's schedule change for the UFSAR commitment, Region I staff felt IP2's progress toward fulfilling the commitment was proceeding in a timely manner and that the schedule change was reasonable.

In June 2000, ConEd provided NRC with a new projected completion date of March 31, 2001, for its commitment to verify the accuracy of the UFSAR, and ConEd reported that it still anticipated completing its DBD initiative by December 31, 2002.

On December 31, 2002, Entergy forwarded correspondence to NRC modifying the completion date for the original commitment that was due on December 31, 2002, to a revised commitment date of December 31, 2003. According to the Region I Administrator and staff, the modification of the completion date was reasonable and acceptable. The Region I Administrator said he considered these deferrals to be appropriate given that numerous, more significant operational and design-related issues emerged over this period requiring extensive licensee management attention and resources.

Region I Oversees IP2 Progress in Fulfilling Design Bases Commitments

According to a Region I Branch Chief, he visited IP2 on April 3, 2001, and verified for himself that the UFSAR update was "essentially done" and that ConEd was "just wrapping up loose ends." The Branch Chief drew this conclusion based on a presentation ConEd gave him describing the methodology for and status of the UFSAR effort. Additionally, he stated that his conclusion was supported by a series of NRC inspections conducted at IP2 since the initial commitment that confirmed progress was being made. OIG reviewed NRC inspection reports from October 1997 through August 2002 and found that some of the NRC inspections specifically looked at the UFSAR and DBD efforts through baseline and special inspections. These reports reflected inspectors' observations that progress continued to be made in these efforts.

The Branch Chief also explained to OIG that when Entergy took over as the licensee for IP2 in September 2001, it assumed ConEd's commitment to complete its DBD Initiative by December 31, 2002, without modifying the completion date. Entergy incorporated the commitment into its "Fundamentals and Improvement Plan" for IP2. With regard to the status of the DBD commitment, the Branch Chief said he visited the plant in May 2002 at which time the plant had completed the review of 22 of the 27 DBDs and planned to complete 3 more by the end of 2002.

According to NRC Inspection Report No. 05-247/2002-010, dated August 28, 2002, which reported results of a supplemental and problem identification and resolution (PI & R) inspection from June 17 through July 19, 2002, Entergy had revised its schedule for completing the DBD effort. According to the inspection report, two remaining DBDs (fire protection and electrical separation) would be completed in 2003, rather than by December 2002. The inspection team concluded that the schedule modification was reasonable.

OIG FINDING

In February 1997, ConEd responded to NRC's 10 CFR 50.54(f) request for information by committing to two separate 24-month efforts at IP2. In the first of these two commitments to NRC, ConEd stated its intent to initiate and complete an UFSAR review program and in its second commitment, ConEd stated it would continue its IP2 DBD Initiative to review and update existing design basis documents and create new ones if needed. Although ConEd initially committed to complete both efforts in 2 years, ConEd revised its projected completion dates two times for the first effort. The UFSAR review program, initially expected to be completed by February 1999, was extended to December 1999, and finally completed by April 2001. The completion date for the second effort was also revised twice, once by ConEd and the second time by Entergy Nuclear Operations, Inc., IP2's current license holder. The DBD Initiative, initially slated for completion by February 1999, was extended to December 2002, and is now expected to be finished by December 31, 2003. OIG found that the NRC staff did not object to the time extensions because it believed each extension was reasonable, given other significant operational problems at the plant, the effort that was required to fulfill the commitments, and the licensee's steady, but slow, progress in addressing them.

II. NRC'S RESPONSE TO REPORTED DISCREPANCIES BETWEEN RPS DESIGN DRAWINGS AND AS-BUILT CONFIGURATION

The Reactor Protection System

The Reactor Protection System (RPS), a system described by NRC staff as "very safety significant" to nuclear power plant operations, is designed to detect a problem in the plant and, if the problem is serious enough, cause the plant to trip (i.e., to automatically shut down in an emergency situation). According to NRC staff, the system can be manually or automatically activated to initiate a plant shutdown. Staff said that to ensure that the reactor will shut down when necessary, the RPS features multiple, independent equipment and components. Any individual RPS component, therefore, could be significant. Furthermore, RPS interfaces with many other safety systems for process monitoring of safety parameters such as reactor coolant pressure, temperature and flow, pressurizer level, steam generator level, and reactor building pressure. As a result, staff said, deficiencies in other systems could have an effect on RPS's ability to operate during an event.

The Region I Administrator told OIG it is a significant problem if the as-built configuration of a system, such as the RPS, is inconsistent with what is needed for the system to be functional. He said it is of lesser significance, but still important, when a system's as-built configuration is inconsistent with design drawings but is still functional. He explained that in either case, inconsistencies between system configurations and design drawings may be indicators that other issues within the system warrant attention.

IP2 Condition Reports Identify Design Bases Discrepancies

OIG learned that in February 2001, a ConEd engineering consultant raised an allegation to Region I pertaining to design bases discrepancies between design drawings and the as-built configuration of the RPS. The allegation referred to 13 IP2 condition reports (CR) that IP2 plant personnel, including the engineering consultant, had written to describe these issues. These CRs were a subset of a larger number (more than 300) of CRs written on RPS between 1998 and 2001.[7] This subset of CRs identified circumstances in which the system's wiring violated statements in the UFSAR. For example, the CRs identified instances of wires associated with computer and alarm circuits being in close proximity of and sometimes in the same cable tray as the wires associated with the trip and logic circuits. The CR reported that these as-built wiring configurations were in conflict with UFSAR wiring separation criteria.

OIG reviewed summaries of the 13 CRs raised in the allegation. Eight of the 13 (CRs 199803574, 199902835, 199903445, 199904968, 200007597, 200009499, 200009641, 200010125) focused on:

- Quality assurance requirements for design verifications,
- Wiring changes resulting from modifications that could not be located, and
- Wiring configurations not in accordance with UFSAR separation requirements.

[7]As context, both regional staff and the IP2 engineering consultant told OIG that roughly 10,000 CRs were being written per year during this timeframe concerning IP2 conditions perceived by licensee staff as in need of attention.

A ninth condition report (CR 200100327) summarized the eight preceding CRs. The remaining four condition reports (CRs 199900478, 199902274, 200008415, and 200008818) documented additional examples of related RPS wiring discrepancies. (See Appendix A for a listing of the 13 CRs and a description of the issues covered in each.)

The engineering consultant told OIG that while employed at IP2 he wrote CR 200100327 as a summary after becoming aware of the eight earlier CRs. These eight CRs summarized documented deficiencies such as wiring separation issues, wiring configurations not in accordance with design drawings, and cable splices not identified on drawings.

The engineering consultant told OIG that he was concerned that collectively these issues warranted a higher level of attention than ConEd had determined was appropriate and that he had raised the matter with ConEd management. Specifically, he explained, he wanted ConEd to perform another Operability Determination (OD) on the RPS to determine whether the system in its current configuration was operable. He told OIG that prior to his writing of CR 200100327, ConEd performed an OD (OD 00-018) on RPS that addressed a subset of the issues raised in CR 200100327. However, he explained that in his opinion that OD did not go far enough to assess the functional changes that may have resulted from the as-found wiring conditions. Dissatisfied with ConEd's response to the issues he raised, and concerned that ConEd would downgrade CR 200100327 from Significance Level (SL) 2 to an SL3,[8] the engineering consultant formally raised the matter to Region I as an allegation.

NRC's Response to RPS Design Bases Discrepancies

OIG reviewed documentation of NRC's response to the issues raised by the engineering consultant and learned that NRC:

(1) inspected several RPS deficiencies prior to the engineering consultant's allegation,

(2) conducted an inspection focused specifically on the RPS wiring discrepancies described in CR 200100327, and

(3) responded directly, in writing, to the engineering consultant on the outcome of NRC's review of the concerns he raised in his allegation.

In the following three sections, OIG describes each of these efforts, which OIG learned, collectively addressed each of the 13 CRs mentioned in the engineering consultant's allegation.

(1) NRC Inspects RPS Deficiencies

OIG learned that prior to receipt of the allegation from the ConEd engineering consultant, and during the course of escalated regulatory activities by Region I subsequent to a steam

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[8]IP2 CRs were ranked on a scale of 1 through 4, with SL1 assigned the highest level of significance. The engineering consultant explained to OIG that CRs assigned a higher SL would receive a more heightened response from ConEd. For example, CRs assigned as SL2 were required to receive a formal Operability Determination, while this was not a requirement for CRs assigned as SL3.

generator tube rupture that occurred at IP2 in February 2000,[9] a team of Region I inspectors conducted a 7-week inspection of "engineering, operations and maintenance, radiation protection, security, and weld radiographs associated with the steam generator replacement project." Inspection activities included a review of a sample of RPS open corrective action items relating to the RPS's nonconformance with design drawings and the UFSAR.

OIG reviewed the inspection report findings pertaining to the RPS review. The inspection report (IR 05-247/2000-014), dated January 2001, described the RPS issue as follows:

> The issue involved the licensee's observation that wiring within the protection racks did not always conform with the statements contained in the UFSAR and electrical separation criteria contained in drawing A208685. Specifically, the licensee found instances of wires associated with computer and alarm circuits being in close proximity of, and sometimes in the same cable tray as, the wires associated with the trip and logic circuits. The licensee also identified examples of switch contacts originally reserved for logic and trip function being used for computer and alarm functions. All potential interactions involved a single train of protection logic and low energy and low voltage circuits.

According to the NRC inspection report, the inspector reviewed three CRs mentioned in the engineering consultant's allegation (CRs 200007597, 200008818, and 200009499) related to RPS logic rack wiring separation concerns, OD 00-018 (dated November 28, 2000), and OD supporting documentation. Based on this review, the report concluded, "There were no significant findings associated with this issue."

The Region I inspector who conducted the review told OIG that the inspection was focused on ensuring that the discrepant conditions reported in the three CRs did not affect the safe operation of the RPS. Although the inspector acknowledged to OIG that it was better to review all open issues and CRs related to a particular system and to sample closed CRs, the inspector explained that he did not do so because of the limited scope of the review coupled with limited manpower resources and time. The Region I Administrator explained to OIG that this sampling of RPS issues was part of a larger review of deficiencies and corrective actions that needed to be addressed at the plant.

(2) NRC Inspects RPS Wiring Discrepancy Issues Described in CR 200100327

OIG learned that following the engineering consultant's allegation pertaining to the RPS, NRC inspectors revisited the issues that the consultant had collectively recorded in CR 200100327 and documented their findings in a June 2001 inspection report (IR 05-247/2001-005) which described the Region I inspectors' review of:

- Corrective actions taken by ConEd to address issues raised in CR 200100327;

[9]On February 15, 2000, IP2 experienced a steam generator tube rupture in one of the plant's four steam generators, which resulted in a minor radiological discharge to the atmosphere.

ConEd's February 12, 2001, OD 01-002, "Ensuring the Functional Capability of a System (RPS) or Component," to determine whether the bases used in the OD were valid and accurate;

Safety Evaluation 99-160-EV to change the UFSAR such that wire separation between safety and non-safety wires was no longer required; and

RPS open condition reports.

These inspection efforts are described below.

Corrective Actions Taken to Address CR 200100327 Issues

OIG reviewed IR 05-247/2001-005, which described Region I's examination of the licensee's corrective actions associated with CR 200100327 and the eight feeder CRs, and corrective actions pertaining to CR 200008415 and one additional CR not referenced in the allegation. The inspection report indicated that as background for the inspection, NRC reviewed CRs 199900478 and 199902274, which had been referenced in the allegation. According to the inspection report, inspectors also:

- Reviewed a ConEd evaluation titled, "SL-2 Evaluation for CR 200100327 on the Reactor Protection System," dated March 7, 2001, to confirm that this evaluation addressed appropriate root causes, contributing causes, compensatory actions and the proposed corrective actions.
- Attended a Corrective Action Review Board (CARB) meeting which reviewed and discussed the evaluation.
- Reviewed the list of ICA (Implementation of Corrective Actions) for CR 200100327 to confirm that the listed corrective actions adequately addressed the root causes and the concerns raised in CR 200100327.
- Reviewed a sample of corrective actions and issues to determine whether these corrective actions were timely and appropriate to address the issues.
- Reviewed the rationale provided for delayed corrective actions.
- Reviewed IP2 documents to confirm that on February 12, 2001, ConEd had generated OD 01-002, "Ensuring the Functional Capability of a System (RPS) or Component," to demonstrate that the RPS can perform its safety function, in spite of the combined wiring and documentation deficiencies.
- Reviewed IP2 documents to confirm that on March 12, 2001, ConEd completed a safety evaluation to address the wiring separation issue regarding RPS wiring configuration conformance with the UFSAR.

Based on this review, the inspectors found no issues that would render the RPS incapable of performing its intended safety function. Specifically, the inspection report stated that no findings of significance were identified.

ConEd's Operability Determination (OD) 01-002, "Ensuring the Functional Capability of a System (RPS) or Component"

According to IR 50-247/2001-005, ConEd generated OD 01-002 "to demonstrate that the RPS could perform its safety function." OIG learned that Region I inspectors reviewed OD 01-002 to determine whether the bases used in the determination were valid and accurate. The inspectors also reviewed supporting documents used in the OD to verify that the data and bases were accurately translated. Supporting documents reviewed included RPS test procedures and test results, a modification for replacing 88 relays in the RPS, and a sample of CRs associated with RPS wiring issues. CRs reviewed included CR 200008818 and two additional CRs not mentioned by the alleger. Based on their review of this issue, the Region I inspectors again concluded that there were "no findings of significance."

Safety Evaluation 99-160-EV

Inspection report 50-247/2001-005 noted that in March 2001, ConEd generated a safety evaluation (SE 99-160-EV) to change the UFSAR so that wire separation between safety and non-safety wires would no longer be required and "safety and non-safety wires can run together within a panduit inside the RPS cabinet." However, according to the Region I inspectors, the safety evaluation did not provide sufficient rationale to justify the change to the UFSAR. According to the inspection report, this matter was not resolved during the inspection and was referred to NRR for review. OIG learned that the results of NRR's review were documented in IR 50-247/2001-010, dated December 17, 2001. In that inspection report, NRR acknowledged that SE 99-160-EV failed to address certain relevant issues; however, NRR concluded that the wiring separation between safety and non-safety wires inside the RPS cabinets was not a design requirement for IP2 and was in compliance with industry standards. Consequently, the wiring configuration at IP2 met design requirements and the issue was closed.

RPS Open Condition Reports

As part of this inspection effort, inspectors also reviewed the RPS condition report history since 1998 and found that since that time more than 300 CRs had been written on the RPS. As of March 9, 2001, 47 CRs remained open in the database, some for almost 3 years. ConEd's records indicated that of the 47 CRs, 3 were ranked as SL4; 37 were ranked as SL3; and 7 were ranked as SL2. The inspection report stated that in response to the inspectors' concerns about possible combined operability or functional effects from the 47 open CRs, ConEd performed an overall assessment of the 47 open CRs and concluded that no functional problems existed. The inspectors reviewed a sample of four CRs to confirm that there were no combined effects that could challenge the functionality of the RPS. The selected CRs were, based on the inspectors' judgement, most likely to yield inspection findings. Based on this review, the inspectors again identified no findings of significance.

(3) Region I Response to Engineering Consultant's Allegation

In a letter dated July 19, 2001, Region I formally responded to the ConEd engineering consultant who wrote CR 200100327 and who subsequently raised the RPS-related issues to Region I. The letter summarized the consultant's RPS-related concerns as presented in CR 200100327, relayed NRC's inspection findings (from IR 50-247/2001-005) pertaining to these concerns, and described the licensee's actions to address them. In its letter to the engineering consultant, Region I addressed the consultant's concern that "there is a lack of response effort and inadequate corrective actions in response to concerns [the consultant] raised regarding deficiencies in the design record and configuration control of the Reactor Protection System (RPS)." The Region I letter also addressed the consultant's concern that OD 00-018 "adequately addressed RPS wire separation and isolation issues, but not the broader concerns" (i.e., loss of design control due to wiring configurations). The letter explained that in response to these concerns, NRC completed an inspection of RPS wiring issues at IP2 on May 4, 2001, that was documented in IR 50-247/2001-005.

The letter also explained that to address the "broader issue for the RPS wiring," ConEd completed an RPS operability determination (OD 01-002) on February 12, 2001, completed a root cause evaluation for CR 200100327, entitled, "SL-2 Evaluation for CR 200100327 on the Reactor Protection System," on March 7, 2001, and established a corrective action program to address other broader aspects of the RPS wiring deficiencies.

In its conclusion to the consultant's concern about RPS configuration control/design record deficiencies, the letter stated,

> . . . your concern was partially substantiated. There were design control weaknesses at IP-2. However, at the time of our inspection, ConEd had established a corrective action plan to address the broader issue as described above [i.e., loss of design control]. Further, our inspection did not uncover any issues that would render the RPS incapable of performing its intended safety function.

The letter also addressed the consultant's concern that CR 200100327, initially assigned an SL of 2, would be reassigned an SL of 3 and that, as a result, ConEd would not conduct an OD "or otherwise address the broader operability issue raised by the lack of quality control in the changes made to the RPS." The letter explained that (1) the licensee did, in fact, complete an OD for the RPS (OD 01-002), which "addressed some important wiring issues;" (2) NRC's inspection did not identify any issues that would affect the functionality of the RPS; and (3) CR 200100327 remained as an SL2 CR.

The Region I inspectors responsible for reviewing the concerns identified by the engineering consultant told OIG that they did not find anything that would render the RPS inoperable.

OIG FINDING

Beginning as early as 1998, ConEd identified problems associated with the IP2 RPS wiring configurations and generated internal CRs to document the findings. These CRs identified circumstances in which the system's wiring violated statements in the UFSAR. Thirteen CRs identifying (or reiterating) such wiring discrepancies were presented formally to the NRC as an allegation by an IP2 engineering consultant who was concerned that collectively the RPS wiring discrepancies warranted a higher level of attention than the licensee had determined was appropriate. OIG learned that Region I performed two inspections relative to these issues and the NRC's Office of Nuclear Reactor Regulation documented its review in a third inspection report. In addition, Region I responded directly to the engineering consultant in a letter dated July 19, 2001. OIG determined that the NRC appropriately responded to the allegations presented to Region I by the engineering consultant. OIG's review of the three inspection reports and Region I's response to the engineering consultant determined that while NRC validated some of the issues the consultant had raised, the agency repeatedly concluded there were no "findings of significance" related to the RPS wiring issues and that ConEd had appropriate measures in place to address the conditions.

III. NRC REGULATORY OVERSIGHT OF IP2'S CAP: 1995 – 2001

Overview of IP2 Operational Problems

Between 1995 and 2001, IP2 experienced a series of operational problems, attributed in part to deficiencies in IP2's corrective action program (CAP) (i.e., its program to self-identify and resolve plant problems). For example,

- In 1995, plant personnel cleaned a turbine using grit. The grit caused significant damage to the internal components of a heater drain tank pump and migrated unchecked throughout the feedwater system, surfacing 2 years later and causing valves to operate erratically.
- NRC inspections conducted between 1996 and 1997 identified various issues, including weaknesses in corrective actions taken to address problems identified by the plant. As a result, in May 1997, NRC issued an NOV citing IP2 for nine violations of NRC requirements, six of which were attributed to corrective action violations.
- In the fall of 1997, IP2 voluntarily shut down to address a large backlog of equipment, programmatic, and performance problems. The plant remained out of service until September 1998.
- In 1998, in NRC Evaluation Team Report 05-247/1998-005, NRC noted that IP2 had identified problems with its CAP in that its corrective action processes were cumbersome and inefficient, many corrective actions were untimely, and completed actions were typically not revisited to determine whether they had achieved their goal.
- In August 1999, IP2 experienced a significant reactor trip, or shutdown, partly due to weaknesses in its CAP.
- In February 2000, IP2 experienced a steam generator tube rupture, also partly attributed to weaknesses in the plant's CAP.
- In May 2000, NRC categorized IP2 as an Agency Focus Plant, a status that denotes a need for increased oversight by NRC.
- In 2001, NRC found that IP2 continued to experience problems in its CAP, including issues pertaining to its RPS.

Significance of the Corrective Action Program

NRC inspects many aspects of nuclear power plants to ensure their safe operation, including the licensees' ability to identify and correct conditions that may affect plant performance and safety. Title 10 of the Code of Federal Regulations, Chapter 50 (10 CFR 50), Appendix B, directs licensees to have a program to assess problems in plant operations and to ensure that timely and effective corrective actions take place. Therefore, it is the licensee's responsibility to implement a program to identify and resolve problems at its facility. Historically this has been referred to as the nuclear power plant's CAP.

NRC Region I staff told OIG that overall plant performance is greatly determined by the effectiveness of a licensee's CAP. Staff told OIG that they expect licensees to be aggressive in identifying concerns and appropriately correcting problems, but they recognize that every plant has problems that need to be addressed. When a CAP is effective, staff said, a licensee is able to identify, prioritize, and quickly resolve conditions that may have a negative impact on plant operations. Staff said they have found that the better performing plants are very aggressive at correcting deficiencies. These plants are also proactive in conducting preventive maintenance and in monitoring plant equipment and conditions. As a result, staff said, those licensees have more durable solutions to their problems than poorer performing plants.

Several staff members interviewed by OIG observed a direct connection between ineffective CAPs and NRC's identification of a plant as an NRC Watch List[10] Plant. According to one staff member, in every case where a plant had problems or became an NRC Watch List Plant, there was a corresponding weakness in the licensee's ability to identify, evaluate, and correct problems, as well as a weakness in assessing the effectiveness of their corrective actions.

The Region I staff told OIG that if NRC lost confidence in a licensee's CAP, the agency would seriously consider whether the licensee should be permitted to operate.

NRC Identifies Repeated Problems With IP2 CAP

OIG was told by the Region I Administrator and staff that between 1995 and 2001, NRC dedicated significant resources to conduct inspections, document findings, and issue sanctions at IP2, yet problems persisted at the plant. Many of the inspections identified problems with IP2's CAP; however, despite heightened levels of NRC attention to these weaknesses, problems related to corrective actions remained unresolved. [See Appendix B for a chronology detailing the significant inspection activity and other oversight efforts performed at IP2 by NRC during this period.]

According to Region I staff, between April 1995 and February 2001, NRC conducted 20 special team inspections at IP2, logging 5,870 inspection hours dedicated to engineering and problem identification and resolution (PI&R).[11] By comparison, the average number of hours devoted to these types of inspections at other single unit[12] Region I nuclear power plants during the same period was 3,854. Furthermore, between 1995 and 2001, IP2 received 13 enforcement actions, 9 of which identified corrective action issues and 8 of which resulted in monetary fines. [See Appendix C for additional information on these 13 enforcement actions.] This expenditure of inspection resources at IP2 was NRC's response to a perceived downward performance trend

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[10]In 1999, there was a change in NRC terminology; Watch List plants are now referred to as Agency Focus Plants.

[11]NRC now refers to the CAP as problem identification and resolution (PI&R). This Event Inquiry, which covers a time period during which the term used to describe the process changed, refers to the process as CAP.

[12]According to a Region I Branch Chief, the term "single unit" generally refers to a nuclear power plant site with only one operating reactor inside the protected area fence. Although there are two operating units at the Indian Point site (IP2 and IP3), Region I treats IP2 as a single unit site due to its past regulatory performance problems. This results in the allocation of more inspection resources at IP2 than would be the case if the plant were treated as a dual-unit site.

that was occurring during the 1995–1999 time frame. According to NRC Region I staff, between 1995 and 2000, overall IP2 performance was not considered very good. Staff said that during that time period, IP2 had problems related to the plant's CAP.

Region I staff told OIG that it viewed 1995 as a downward turning point for the plant and recalled the grit intrusion event that occurred that year as an example of this decline. Between October 1996 and April 1997, NRC staff conducted four inspections of IP2, which resulted in the issuance of an NOV in May 1997 based on nine violations of NRC requirements. The inspections included an Integrated Performance Assessment Process (IPAP) and three routine inspections conducted by the NRC resident inspectors. Problems identified during the inspections included weaknesses in IP2's design control which, staff explained, pertained to the availability and completeness of design bases information and problems with the CAP.

The Region I Administrator told OIG that following February 1997 there was a series of events that occurred at IP2, coupled with NRC's inspection findings, that reinforced his concerns about IP2's declining performance. He told OIG that the NRC subsequently sent a message to ConEd management by issuing fairly significant civil penalties and a confirmatory action letter (CAL).[13] Additionally, he met with ConEd's Chief Executive Officer to address NRC's concerns about IP2's declining performance, the decline in overall effectiveness of management oversight, and a perception that management tolerated problems rather than aggressively identifying and correcting them.

Consequently, ConEd management responded to Region I by documenting actions it planned to take to arrest the performance declines and to improve the quality of these activities. These detailed action plans were included in a program that ConEd identified as the *Strategic Improvement Program*.

Declining Systematic Assessment of Licensee Performance Scores

According to the Region I Administrator and Region I staff, the Region's concerns about IP2's performance in 1996 were documented in NRC's Systematic Assessment of Licensee Performance (SALP) scores and periodic SALP reports for IP2. The SALP was an NRC evaluation of plant performance conducted every 12 to 24 months within the parameters of NRC's inspection program. The report included a numerical rating of the plant in four categories — plant operations, maintenance, engineering, and plant support — as well as a narrative discussion of performance in each area.

In the SALP report covering the period from September 17, 1995, through February 15, 1997, Region I staff noted that overall performance at IP2 declined. Performance in the areas of operations and plant support were rated as generally effective and some elements were very good; however, performance declined in maintenance and substantively declined in engineering. The SALP report noted many equipment problems were due to the poor condition

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[13]CALs are letters issued by NRC to licensees or vendors to emphasize and confirm a licensee's or vendor's agreement to take certain actions in response to specific issues.

of a number of systems. Licensee management was involved in many plant activities and made operational decisions, but management oversight was at times ineffective regarding overall efforts to identify, evaluate, and correct problems.

IP2 Shuts Down To Address Backlog of Problems

OIG learned that following repetitive failures of safety-related electrical breakers, IP2 voluntarily shut down to address a large backlog of equipment, programmatic, and performance problems. This outage lasted from October 1997 until September 1998. According to NRC staff, IP2 used this period to try to better identify and correct these deficient conditions at the plant.

Instead of conducting a planned Operational Safety Team Inspection (OSTI)[14] of IP2, NRC permitted ConEd to hire a team of independent experts to conduct an Independent Safety Assessment (ISA) of the power plant in the spring of 1998. NRC assembled a special NRC Evaluation Team (NET) to gauge the validity and effectiveness of the ISA and review the outcome. The NET observed and evaluated the IP2 ISA from March 30 through May 7, 1998, to assess the validity of the ISA conclusions and to determine whether the ISA had fulfilled NRC's intent to obtain an OSTI-type performance assessment. According to the NET report, the ISA achieved noteworthy insights, including the identification of problems with IP2's CAP. Specifically, the ISA found that the CAP was cumbersome and inefficient, many corrective actions were untimely, and completed actions were typically not revisited to see whether they had achieved their intended impact. According to an NRC staff member who participated in the review, IP2's CAP "was not working very well at all."

Subsequent to the ISA findings, ConEd developed plans to improve station performance and, according to the regional staff and inspection reports, IP2's performance began slowly to improve following plant startup in September 1998. According to the NRC staff, inspection reports, and other docketed correspondence between NRC and ConEd, substantial changes were made to IP2's CAP during this period. However, although progress was made, a number of problems remained that required continued licensee management attention.

IP2 Experiences Two Significant Events

In August 1999, IP2 experienced a reactor trip, or shutdown, a risk-significant event that NRC staff characterized as preventable and partly attributable to weaknesses in IP2's CAP. The reactor trip was caused partly by a condition involving repetitive problems with one channel of RPS's over-temperature/delta-temperature circuitry. The condition, which existed since January 1999, had not been promptly identified, the cause of the condition had not been determined, and corrective actions had not been taken. According to the Region I Administrator, while the August 1999 event challenged safe operation, safety margins were maintained at an acceptable level.

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[14]At the time, OSTIs were conducted to supplement normal inspections for special purposes such as to verify that a plant operator has properly prepared the staff and the plant for resumption of power operations after an extended shutdown. These inspections were performed by either a headquarters or regional team and typically consist of a 2-week onsite inspection conducted by a team of seven inspectors and a team leader.

In February 2000, IP2 experienced yet another significant problem attributed to weaknesses in IP2's CAP: a steam generator tube ruptured in one of its four steam generators, resulting in a leak that allowed pressurized radioactive water, which acts to cool the reactor, to mix with non-radioactive water in the steam generator. The power plant was manually shut down following the event. This resulted in a minor radiological discharge to the atmosphere.

CAP Problems Persist at IP2

In an NRC inspection report (IR 05-247/2001-002) issued in 2001, the Region I inspection team again noted weaknesses in IP2's CAP. According to the report, IP2's progress to effect change continued to be slow. The report "noted problems similar to those that have been previously identified at the IP2 facility, including those in the areas of design control, human and equipment performance, PI&R, and emergency preparedness."

When interviewed by OIG, Region I staff attributed IP2's CAP problems to a large backlog of problems — any one of which might not appear significant. Staff said that IP2 was able to identify problems but was frequently ineffective at prioritizing and correcting them and determining their root cause. Staff attributed this specifically to a cultural problem at IP2 that was not recognized by ConEd management until after the August 1999 event. Staff described this culture as one in which ConEd management did not emphasize or encourage staff efforts to prioritize the correction of problems and identify root causes.

The Region I Administrator and staff acknowledged that the improvements at IP2 were slow, and in some respects limited, but steady. The Region I Administrator told OIG that IP2 met NRC's minimum regulatory requirements and there was never a situation where the margins of safety had been reduced to a point where the plant was unsafe. He added that as a regulator one has to work within the regulatory framework and distinguish between conditions that are unsafe and conditions that involve weaknesses in performance. The Region I Administrator and staff repeatedly emphasized to OIG that the increased inspections and aggressive oversight never identified a situation where IP2 was unsafe.

The Region I Administrator explained to OIG that IP2's rate of improvement above fundamental protection of public health and safety is determined by the plant management. The licensee determines the type and amount of resources that it will apply to facilitate improved performance. The licensee also makes personnel selections at the plant and it is ultimately up to the individuals hired to make these improvements and effect change. The Region I Administrator told OIG that he continually pressed ConEd management to strengthen the margins of safety at IP2 by conducting numerous inspections and special assessments and by communicating the Region's findings to ConEd in a clear and direct manner.

OIG FINDING

Between 1995 and 2001, IP2 experienced a series of operational problems, attributed in large part to deficiencies in IP2's CAP. OIG found that during this period, Region I dedicated significant resources to conduct inspections, document findings, and issue sanctions, yet problems persisted at the plant. Between April 1995 and February 2001, NRC conducted 20 special team inspections at IP2, logging 5,870 inspection hours dedicated to engineering and problem identification and resolution. Furthermore, between 1995 and 2001, Region I issued 13 enforcement actions to IP2. Many of the inspections identified problems with IP2's CAP. However, despite heightened levels of NRC attention to these weaknesses, problems at IP2 remained unresolved. OIG found that in spite of the intensified regulatory oversight by Region I, IP2 was only able to achieve limited improvement in plant performance.

IV. NRC'S UTILIZATION OF THE SMM PROCESS TO HEIGHTEN ATTENTION AT IP2

Senior Management Meeting Process

Between 1986 and 2001, NRC held Senior Management Meetings (SMM) semiannually to allow NRC senior managers to focus agency attention on those plants of highest concern and to monitor licensee efforts to recognize and resolve performance problems. According to the March 1997 version of NRC Management Directive (MD) 8.14, "Senior Management Meeting (SMM)," the primary goal of an SMM was to identify declining trends in the operational safety of individual plants so that early corrective actions could be implemented. OIG was told by senior NRC managers that the SMM offered a means to communicate NRC's concerns to licensees with poor or adverse performance trends.

During the SMM, the senior NRC managers could opt not to take action regarding a particular plant or they could choose to take one of several actions to heighten oversight. For example, senior managers could choose to issue a Trending Letter to advise a licensee that NRC had taken notice of declining plant performance and that if performance did not improve, the plant might be placed on the NRC's Watch List. Or, the managers could choose to place a plant directly on the Watch List. A plant placed on the Watch List received increased oversight from NRC in the form of additional inspections, letters expressing agency concerns about declining performance, and other types of regulatory attention. According to the NRC staff, designation as a Watch List plant could also bring significant public attention to a licensee and could result in a negative economic impact for the utility. These potential negative consequences would motivate a licensee to improve plant performance.

Senior Management Meetings were chaired by the NRC Executive Director for Operations. Participants typically included the Deputy Executive Director for Regulatory Programs; Deputy Executive Director for Regulatory Effectiveness, Program Oversight, Investigations and Enforcement; Deputy Executive Director for Management Services; Regional Administrators; Directors of the Offices of Nuclear Reactor Regulation, Analysis and Evaluation of Operational Data, Nuclear Material Safety and Safeguards, Nuclear Regulatory Research, Enforcement , Investigation, and State Programs; and senior managers from the Office of the General Counsel.

Region I Administrator Seeks SMM Action on IP2

OIG learned that paralleling NRC's inspection activity at IP2 from 1997 through 2000 was a series of attempts by the Region I Administrator to further heighten NRC oversight at the plant through the agency's SMM process. At the June 1997 SMM, the Region I Administrator presented his concerns regarding the declining performance of IP2 that was the result of significant equipment, human performance, and technical support performance issues that were apparent in late 1996. NRC Regional Administrators and senior managers told OIG that at the June 1997 SMM, the Region I Administrator made "a strong presentation" regarding IP2's performance and his belief that IP2 should be designated as a Watch List plant. However, the senior managers decided not to designate IP2 as a Watch List plant but to continue providing the heightened level of regional oversight underway at the time. According to the senior managers, and based on minutes of the SMM proceedings, the information presented at the SMM did not identify a situation where the plant was unsafe, a safety system was inoperable, or adverse trends were apparent. Thus, the senior managers determined that IP2 did not warrant agency-level action.

During the SMM held in January 1998, the Region I Administrator again presented IP2 for discussion asserting that there had been little change in performance in most respects over the prior 6 months; that recent inspections raised additional concerns with respect to performance; that NRC inspectors, rather than ConEd, continued to identify many of the performance problems, particularly in operations and engineering; and that equipment and human performance issues continued to be of concern. Additionally, the informality of processes contributed to problems observed in several areas, including technical specification implementation, procedural adherence, problem identification, and timely effective resolution of issues. OIG learned that this time, the consensus of the senior managers was to conduct a diagnostic-type review to obtain additional information on the plant's condition and not to issue a trending letter or put the plant on the Watch List. Again, the senior managers believed that Region I did not identify a situation where the plant was unsafe or a safety system was inoperable; however, they acknowledged that IP2 continued to exhibit performance weaknesses, and they noted that a definitive improvement trend was not apparent.

In July 1998, the Region I Administrator again presented IP2 at the SMM in the belief it should be designated as a Watch List Plant. He asserted that the performance at IP2 was largely unchanged during the preceding 6 months with respect to human performance and the control of plant activities. Additionally, the 1998 Independent Safety Assessment (ISA) conducted by ConEd identified some important deficiencies and weaknesses that existed at IP2 particularly in the areas of management and operations. Despite the Region I Administrator's presentation, the SMM again declined to designate IP2 a Watch List plant. This time, the SMM decided to maintain, rather than increase, the level of attention to allow the licensee a period of time to execute its performance improvement initiatives. The senior managers recognized that IP2 continued to have performance weaknesses, but again they believed that Region I did not identify a situation where the plant was unsafe or a safety system inoperable.

IP2 was not discussed during the April 1999 SMM. The Region I Administrator told OIG that he did not recommend that IP2 be presented for discussion because it had experienced no significant events since the last time he presented the plant for SMM discussion. He felt that in 1999, performance weaknesses still existed but that IP2 was no worse than in preceding years and was, in fact, slowly improving. He said he still would have preferred SMM action; however, he felt he lacked a basis for presenting the plant at the SMM.

SMM Designates IP2 as Agency Focus Plant in May 2000

In May 2000, the Region I Administrator presented IP2 at the SMM after the occurrence of two significant events at the plant, the August 1999 reactor trip and the February 2000 steam generator tube rupture. OIG learned that overall, the events and related findings during this assessment period represented issues that were of substantial significance; therefore, the senior managers categorized IP2 as an Agency Focus Plant under the revised SMM process.[15]

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[15]In April 1999, the Commission approved SECY 99-086, "Recommendations Regarding the Senior Management Meeting Process and Ongoing Improvements to Existing Licensee Performance Assessment Processes." SECY 99-086 eliminated the "Watch List" and proposed that during SMM meetings, participants would

According to the SMM minutes, the senior managers concluded that the broad performance issues that had existed at IP2 for the past several years revealed a number of deficiencies in the plant's CAP and that IP2 improvement initiatives yielded some progress but, overall, were limited in remedying the underlying problems.

According to the Region I Administrator, the August 1999 and February 2000 events revealed the depth of IP2's performance problems and were evidence of the significant issues discussed at previous SMMs. Region I staff echoed this sentiment to the OIG, questioning why — given the inspection history, the identified problems, the NRC man-hours at the plant, and the history of civil penalties — IP2 was not put on the Watch List sooner.

Current Status of IP2

Region I staff has informed OIG that since March 2001, NRC has provided a significant amount of oversight and inspection effort at IP2. The Region I staff performed 12,950 hours of inspection activity at IP2 between March 1, 2001, and March 1, 2003, compared to an average of 8,297 hours at other single unit sites in Region I. (See Appendix B for a chronology of NRC inspection activity at IP2 during this time period.) Of the 12,950 hours of inspection performed at IP2 during this 2-year period, 2,216 hours were focused on engineering and PI&R compared to an average of 1,077 hours devoted to these areas at other single-unit Region I sites. The staff informed OIG that these figures indicate that during this period, IP2 has received about 1.5 times as much inspection as the average for other single-unit sites and about 2 times as much inspection pertaining to engineering and PI&R.

Annual assessments of plant performance[16] performed since the plant was categorized as an Agency Focus Plant in May 2000 indicate that IP2 performance has been improving, albeit slowly, since that time. NRC's annual assessment of plant performance for April 2, 2000, to March 31, 2001, found that while IP2 met all cornerstone objectives, it remained in the Multiple/Repetitive Degraded Cornerstone column of the NRC's ROP Action Matrix. According to the Region I staff, that assessment noted a number of issues in design control, equipment reliability, PI&R, and human performance. While some performance improvements were noted, progress was considered slow and limited in some areas. Region I staff noted that as of December 31, 2001, IP2 remained in the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix.

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determine whether a plant warranted Agency Focus (characterized by NRC Executive Director for Operations and Commission involvement, e.g., issuance of an order), Regional Focus (managed by the regional administrator, e.g., issuance of a confirmatory action letter), or routine oversight.

[16]Under the ROP, NRC assesses licensee performance in various ways, including quarterly plant performance assessments based on inspection findings and performance indicator data. Regional offices conduct a more comprehensive review after the second quarter of the year (mid-cycle) to assist in planning inspections for the next 6 to 12 months. The regions also conduct an annual (end-of-cycle) review after the fourth quarter of the year to develop an annual performance summary for each plant and to plan inspections for the next 12 months. NRC uses an Action Matrix to assist staff in reaching objective conclusions regarding licensees' safety performance. The matrix allows for plants to be categorized into five possible results categories, or matrix columns, which indicate the plant's level of performance and the agency's required response. Categories (from lowest to highest performance) are (1) Unacceptable Performance, (2) Multiple/Repetitive Degraded Cornerstone, (3) Degraded Cornerstone, (4) Regulatory Response, and (5) Licensee Response.

Significant inspection activity continued during 2002, including an augmented PI&R inspection and supplemental team inspection in June and July 2002. OIG was told that in August 2002, IP2 had made sufficient progress to justify removal of the plant from the Multiple/Repetitive Degraded cornerstone into the Degraded Cornerstone column of the Action Matrix. OIG was told by the Region I Administrator that on February 7, 2003, NRC completed its end-of-cycle plant performance assessment of IP2 covering performance from January 1, 2002, through December 28, 2002. NRC concluded that during that time period, IP2 continued to operate in a manner that preserved public health and safety.

The Region I Administrator and staff told OIG that Region I fully utilized the regulatory tools it had available to deal with IP2. The Region I Administrator said that although the plant was never unsafe, improvement in IP2's performance might have been swifter had the plant been designated a "Watch List" plant by the SMM earlier. This designation would have sent a powerful message to the licensee concerning the need for improved performance.

The Region I Administrator commented that while the agency's senior managers designated the plant as an "Agency Focus Plant" in May 2000, this occurred after the plant had reversed its downward trend and, in fact, the designation had a relatively small impact on recent plant operations because the plant's declining performance had already been arrested as a result of earlier actions taken by the NRC. The Region I Administrator also noted that SMM deliberations were always thorough but that decisions were inherently difficult given the complexity of issues involved.

Additionally, the Region I Administrator commented to OIG that Entergy's purchase of IP2 in September 2001, had a considerable impact on plant performance. According to the Region I Administrator, Entergy conducted its own self-assessment of IP2 and subsequently committed significant resources to the plant. Furthermore, Entergy had experience operating other nuclear power plants, was aware of the need to inject resources to improve plant performance, and had those resources available. Entergy also understood the need to bring top management talent to operate the plant, which it did. According to the Region I Administrator, this shift in ownership facilitated the IP2 improved performance trend.

The Region I Administrator considered IP2's improvement as an NRC "regulatory success story." He stated that NRC's aggressive oversight and intervention arrested the decline in early 1996 and prevented IP2 from ever getting to the point where it was unsafe to operate. He acknowledged that IP2's improvement has been slow at times and often uneven, but that, overall, plant performance has steadily improved. In his view, the conditions that led to IP2's poor performance in the mid-1990s developed over a number of years and, therefore, required time to resolve. He credited NRC oversight efforts performed at IP2 since 1996 with having caused the plant to reverse its downward performance trend and begin its slow progress toward the performance improvement reflected in the NRC's recent assessment letters.

OIG FINDING

On four occasions between 1997 and 2000, the Region I Administrator sought additional NRC oversight for IP2 by seeking to have NRC's senior managers place IP2 on NRC's Watch List via the agency's Senior Management Meeting process. However, it was not until May 2000, after the August 1999 reactor trip and the February 2000 steam generator tube rupture, that NRC senior managers agreed that this form of heightened attention was appropriate. In May 2000, IP2 was classified as an Agency Focus Plant. Subsequent to being so designated, NRC annual assessments of plant performance indicated that IP2 had improved. OIG concurs with the Region I Administrator and his staff that placing IP2 on the Watch List sooner might have sufficiently motivated the licensee to cause earlier improved performance.

APPENDIX A
Summary of IP2 RPS Condition Reports

CR 199803574 identified a discrepancy between the RPS wiring configuration and a description in section 7.2.2.9 of the UFSAR of isolation between safety signals and annunciator and/or computer signals. Contrary to the UFSAR statement that "The center and front decks of RPS logic relays are used for annunciator and computer signals respectively," 22 RPS logic relays were found to violate this criterion.

CR 199900478 identified discrepancies between design drawings and the as-built configuration with respect to contact state associated with interposing relays for the low autostop oil pressure protection scheme. The corrective action for this condition involved revision of four drawings to reflect the field condition.

CR 199902274 identified "minor" inconsistencies affecting 14 RPS and ESF drawings. Corrective action involved revising the affected drawing based on comments received from an outside contractor who was tasked with the drawing review.

CR 199902835 identified three distinct discrepancies between plant drawings and the as-built condition. These discrepancies involved: RPS logic relays used to block the "Source Range High Influx at Shutdown" annunciator, drawings showing RPS relay contact configuration different from the as-built condition, and incorrect RPS relay nomenclature on plant drawings. The corrective action for this CR was limited to revising the affected drawings to agree with the as-found condition.

CR 199903445 was initiated because the drawing revisions prepared in response to CR 199902835 were in error. This CR also identified an additional drawing error in which the drawing showed the incorrect RPS relay contacts used for the Source Range High Flux at Shutdown annunciator block.

CR 199904968 identified another discrepancy between the design drawings and the as-found configuration of the RPS. This discrepancy involved contacts from RPS relay P10-2 that are used to defeat the Source Range Loss of Detector Voltage annunciator above 10% reactor power which are not shown on plant drawings. The corrective action for this CR involved a field verification of the configuration and revision of the affected drawing to reflect the as-found condition.

CR 200007597 identified a number of potential internal wiring related discrepant conditions in the reactor protection racks. Isolated cases of wire routing and/or terminations were observed to be inconsistent with routing/separation requirements stated in the UFSAR. In response to this CR an Operability Determination (OD) 00-018 was issued to address the wiring routing/separation issues. The OD determined that the RPS was operable.

CR 200008415 identified drawing discrepancies between Westinghouse RPS wire lists and field conditions, however, an operability determination concluded that this did not constitute an operability concern.

CR 200008818 identified a broken contact in a reactor trip relay, unidentified, unterminated switchboard wire with exposed lugs in RPS cabinets, and a mixing of wiring associated with computer/logic/annunciator functions. The broken contact has been repaired. A 200-degree hold was placed on this CR. The "Operability Review Note" by the Watch Engineer stated "200 degree hold for loose wires, etc." The response to the unterminated (loose) wire issue was not addressed. The engineer who responded to the 200H action stated that he considered the unterminated wire a housekeeping issue and therefore, did not address it as part of the 200H response.

CR 200009499 identified additional conditions in which the wiring in the RPS racks violated statements in the UFSAR. The CR stated that "Wires (in RPS Racks 4 and 5) were carelessly strewn through multiple wire ways," and "Had the original design been followed, there would have been no mixing (of circuit functions) and there would have also been half as many new wires to mix." These issues were addressed in Operability Determination 00-018 which was conducted on CR 200007597 which found that the RPS was operable.

CR 200009641 identified six issues related to RPS wiring deficiencies or discrepancies, three of which were similar to or a repeat of issues identified in previous CRs. The new issues included a wire associated with an NIS power range logic relay with a splice that is not represented on plant drawings and single cable containing both 125 VDC logic protection power and 118 VAC instrument bus power. Both of these issues were addressed in Operability Determination 00-018.

CR 200010125 identified discrepancies between design drawings and the as-built configuration of the RPS. This CR also identified other CRs that described similar inconsistencies between design drawings and RPS wiring. A review of the corrective action associated with these CRs revealed that the CR actions were typically closed by revising the plant drawings to reflect the as-found configuration without performing a safety evaluation to determine the impact of the change on the design and licensing basis. In some cases the as-found condition affected the system design as depicted in the UFSAR text and/or figures. This CR also identified errors made in drawings as part of the corrective action for CR 199904968. Furthermore, this CR identified discrepancies between drawings and the as-found RPS wiring that had not been previously identified.

CR 200100327 summarized numerous issues identified in eight previously submitted CRs that documented a lack of configuration control and quality control of changes to the RPS wiring since 1998. The concerns raised in CR 200100327 were categorized as quality assurance requirements for design verifications, wiring changes resulting from modifications that could not be located and wiring separation not in accordance with the UFSAR. The eight CRs summarized in CR 200100327 are CR 200010125, CR 199803574, CR 199904968, CR 199902835, CR 199903445, CR 200007597, CR 200009499 and CR 200009641.

APPENDIX B
Chronology of Significant Inspections and Oversight at IP2, 1995 – 2003[1]

March 14, 1995	Inspection Report (IR) 1995-01, special safety inspection of AFW digital controller failure.
April 12, 1995	IR 1994-017 service water self-assessment inspection.
August 28, 1995	IR 1995-080, Operational Safety Team Inspection.
October 26, 1995	SALP report issued.
January 28, 1997	IR 1996-080, Integrated Performance Assessment Process (IPAP).
January 31, 1997	Confirmatory Action Letter (CAL) issued.
February 21, 1997	CAL closed.
March 31, 1997	Final SALP report issued.
May 1, 1997	Plant shutdown for refueling outage.
May 9, 1997	IR 1997-003 integrated inspection.
June 19, 1997	IR 1997-005, special inspection conducted for stuck open MSSV.
June 1997	IP2 discussed at Senior Management Meeting (SMM).
June 1997	Regional Administrator meets with ConEd Chief Executive Officer.
July 8, 1997	Plant startup from refuel outage.
July 26, 1997	Generator load rejection and reactor trip.
July 28, 1997	Reactor trip.
August 6, 1997	Shutdown.
August 8, 1997	IR 1997-008, special inspection of outage issues.
August 23, 1997	Reactor trip due to reactor coolant pump breaker testing logic error.
August 25, 1997	Plant startup.
September 29, 1997	IR 1997-010, special inspection of load reject and reactor trip.

[1]Information in this chronology was provided to OIG by Region I.

October 14, 1997	Plant shut down due to repetitive DB50 circuit breaker failures.
December 12, 1997	IR 1997-012, integrated inspection report, resident inspection and specialist review of safety-related breaker problems.
January 1998	IP2 discussed at SMM.
January 1998	Performance letter issued to ConEd - decision made to perform Operational Safety Team Inspection/Independent Safety Assessment.
February 13, 1998	IR 1997-013, special inspection of 480 Vac Breaker failures.
March 26, 1998	CAL 1-98-005 due to issues discovered during shut down not related to circuit breakers.
March 26, 1998	IR 1998-201, design inspection.
April 27, 1998	NRC restart action plan for IP2 issued.
May 1998	Independent Safety Assessment performed by ConEd.
June 3, 1998	IR 1998-005, NRC Evaluation Team (NET).
June 26, 1998	IR 1998-006, special inspection focusing on corrective actions regarding plant restart issues.
June 1998	Emergency preparedness exercise.
July 9, 1998	Revised NRC CAL 1-98-005 issued March 26, 1998.
July 1998	IP2 discussed at SMM.
September 16, 1998	IR 1998-012, followup NRC NET evaluation team inspection.
September 21, 1998	Reactor startup.
October 16, 1998	IR 1998-008, special Inspection of corrective action associated with restart issue.
October 23, 1998	IR 1998-014, NRC integrated inspection.
November 3, 1998	IR 1998-016, NRC special inspection of high efficiency particulate air (HEPA) filter deterioration.
January 29, 1999	IR 1998-018 NRC 40500 Corrective Action Program Inspection.
April 1999	Plant Performance Review.

June 1999	ConEd external assessments of operations, work control, and maintenance departments.
August 19, 1999	IR 1999-004 NRC team inspection report (Core Engineering Team).
August 31, 1999	Reactor trip and loss of offsite power.
September 14, 1999	Management meeting - Augmented Inspection Team (AIT) interim results.
September 23, 1999	Public exit meeting - AIT exit meeting.
September 1999	Emergency preparedness exercise.
October 13, 1999	Reactor startup.
October 19, 1999	IR 1999-008, AIT.
October 1999	IR 1999-013, AIT follow up team inspection commenced.
October 1999	Mid-cycle plant performance review letter issued.
November 23, 1999	Public Meeting - IP2 performance assessment results from September 1999 plant performance review.
December 21, 1999	Results of the follow-up inspection to the AIT (1999-013).
December 1999	IP2 Recovery Plan actions transferred to Business Plan.
January 5, 2000	IR 1999-014. Results of enforcement follow up of AIT for August 31, 1999 trip.
January 7, 2000	Drafted charter for the formation of the Indian Point Unit 2 oversight panel (IPOP).
February 1, 2002	Drafted IP2 oversight strategy.
February 15, 2000	Reactor trip - steam generator tube failure (SGTF).
March 1, 2000	SGTF meeting.
March 14, 2000	SGTF public meeting.
March 2000	Formation of IP2 communications team.
March 2000	Plant performance review letter.
April 28, 2000	NRC AIT SGTF IR 2000-002 issued.

May 23, 2000	IP2 discussed at SMM; letter issued characterizing IP2 as an "Agency Focus" plant.
June 25, 2000	Public meeting.
July 10, 2000	IR 2000-007, AIT SGTF follow-up.
July 27, 2000	IR 2000-010, NRC SGTF special inspection.
August 3-4, 2000	Regional Administrator site visit.
August 31, 2000	IR 2000-010, SGTF special inspection.
September 11, 2000	NRC Agency Focus Meeting. (Regional Administrator and NRR Deputy Director Site Visit)
September 26, 2000	Regulatory conference on SGTF "red" finding.
September 2000	Ongoing regional management briefings on cornerstone deficiencies, and plant performance issues throughout restart.
October 2, 16, 2000	Problem Identification and Resolution (PI&R) inspection.
October 5, 2000	EDO brief to discuss content of "Agency Focus" letter.
October 10, 2000	Assessment follow up (Agency Focus Update) letter.
October 11, 2000	ROP meeting held in Cortland Town Hall.
October 16, 2000	Operator requalification Inspection.
October 25, 2000	NRC - ConEd management meeting.
October 31, 2000	Significant Determination Process repanel (final determination of "red or yellow" finding for SGTF issues).
November 1, 2000	IP2 SGTF Lessons Learned Task Force (LLTF) report issued.
November 6, 2000	NRC on-site restart readiness reviews.
November 8, 2000	Mid Cycle review meeting conducted.
November 14, 2000	RI review of four system readiness reviews.
November 16, 2000	Public meeting.
November 16, 2000	NRC noted that the independent 125 VDC SSFA team performed a high quality review.

November 20, 2000	Issued red finding and Notice of Violation (NOV) for the poor SG inspection program that led to the SGTF.
November 27, 2000	NRC safety system readiness review inspection on the Safety Inspection system.
November 29, 2000	Mid cycle performance review and inspection plan letter issued.
December 1, 2000	Region I senior management site visit to IP2.
December 4, 2000	PI&R inspection report.
December 6, 2000	EDO briefing.
December 11, 2000	Plant heat up above 200 degrees - restart inspection begun.
December 18, 2000	IR 2000-014 design issues inspection.
December 20, 2000	NRC replied to ConEd's request for extension to respond to the red finding and NOV.
December 22, 2000	NRC Region I issues NRC review efforts/status letter.
December 30, 2000	Plant restarted.
January 2, 2001	Turbine trip due to low SG level.
January 5, 2001	Regional Administrator visits Congresswoman Kelly.
February 9, 2001	95003 multiple degraded cornerstone supplemental inspection.
February 26 - May 4, 2001	IR 2001-005, review reactor protection system (RPS) design issues.
February 27, 2001	Chilling effect letter issues.
March 1-2, 2001	Regional Administrator site visit and public exit meeting for 95003 inspection.
March 9, 2001	Chairman site visit with Regional Administrator and Executive Director for Operations.
April 3, 2001	Division of Reactor Safety (DRS) branch chief visit to IP2 - UFSAR verification project status.
April 10, 2001	IR 2001-002, (95003 Inspection) supplemental inspection report issued.

June 18, 2001	IR 2001-007, emergency preparedness (EP) exercise review and supplemental inspection of licensee actions to address three findings in the EP cornerstone area.
July 23, 2001	IR 2001-007, review of 2001 design engineering business plan and scope and 50.54 (f) commitment status.
July 23, 2001	IR 2001008, review of 2001 Design Engineering Business Plan Scope and 50.54(f) commitment status.
October 22, 2001	IR 2002013, NRC on-site to do initial inspection of the failure of three of six crews on licensed operator (LOR) examinations and to observe facility evaluate seventh crew; crew fails: four of seven = yellow finding.
November 5, 2001	IR 2001-010, review of licensee's safety injection (SI) safety system functional assessment (SSFA) and PI&R inspection.
November 27, 2001	IR 2001-011, NRC observes facility-led evaluation of an operating crew; while onsite, conducts regular-hours control room (CR) observations.
December 7, 2001	IR 2001-011, NRC- led evaluation of another operating crew; while onsite, conducts regular-hours CR observations.
December 16, 2001	IR 2001-011, NRC- led evaluation of 4 staff RO licenses.
January 28, 2002	IR 2001-014, review of licensee's self assessment and Fundamentals Improvement Plan (FIP), including the Design Basis Initiative (DBI).
February 7, 2002	IR 2002-007, NRC observes facility-administered evaluations (High Intensity Training (HIT).
March 21, 2002	IR 2002-007, NRC observes facility-administered evaluations (HIT).
March 21, 2002	IR 2002-009, supplemental inspection to review causes and corrective actions for yellow finding related to operator requalification.
June 24, 2002	IR 2002-010, augmented PI&R inspection, reviewed performance issues related to the multiple degraded cornerstone designation, progress implementing the FIP, and review of the degraded control room west wall fire barrier.
November 4, 2002	IR 2002-007, review of reactor protection system (RPS) wiring verification.
December 9, 2002	IR 2003-002, PI&R team inspection.
December 2002 - February 2003	IR 2003-003 and IR 2003-005 (both draft), team inspections to review TI 2515/148 and various other security issues.

January 27, 2003	IR 2003-004 (draft), engineering team inspection reviewed design and performance capability of component cooling water and offsite power supplies.

APPENDIX C
Summary of Escalated Enforcement Action from 1995-2000

1996-01, Enforcement Action 96-089, Significance Level (SL) III
10 CFR 50.59 (SL III) and 50.72 (SL IV)

Repair activities on central control room roof left ventilation system in unanalyzed condition for 2 months. Inadequate corrective actions.

1996-04, Enforcement Action 96-272, SL IV
Criterion XVI (SL IV) and Technical Specification (TS) 6.8.1. (SL IV)

1) Failure to maintain proper configuration control over containment isolation valve, contrary to procedure requirements.
2) Failure to preform required safety evaluation on procedure change.

1996-07, Enforcement Action 97-031, SL III ($50,000 civil penalty)
Criterion XVI (SL III)

Inadequate measures were taken to assure that the cause of each condition was determined and corrective action taken to preclude repetition.

1) Repeated surveillance test failures associated with the TDAFW pump's steam admission valve and discharge flow control valves. Valve damage subsequently identified.
2) Preconditioning of TDAFW pump by blowing down steam traps prior to testing. Adequate engineering review was not performed to support pump operability.
3) Multiple surveillance test failures associated with alternate safe shutdown system power transfer switches for the 23 and 24 service water pumps.
4) Untimely identification of degradation of PAB filter/fire deluge system control panel and associated circuits. System was incapable of performing design function. Poor implementation of an alarm response procedure's required actions.

1996-08, Enforcement Action 97-113, SL III ($50,000 civil penalty)
Criterion XVI (SL III), TS 6.8.1(SL IV), TS 6.5.1.6.a. (SL IV)

1) Failure to take adequate corrective actions following grit intrusion during the 1995 refueling outage. Resulted in inoperability of three of the four safety-related MFRV's and one low-flow bypass MFRV in January 1997.
2) Control of SG levels not in accordance with procedure and the failure to make temporary procedure changes to invoke administrative allowances for situation where deviation is necessary.
3) Failure to perform a required review of a vendor report that was used as the basis to support DG operability following the 1995 grit intrusion.

1996-80, Enforcement Action 96-509, SL III (\$50,000 civil penalty)
Appendix R (SL III)

Fire protection features not provided to protect one train of systems - two instances.

1) Certain normal safe shut down instrumentation and the corresponding alternate safe shutdown instrumentation would be subject to fire damage.
2) Potential for hot shorts exists as a result of fire damage to cables associated with both the pressurizer PORV and block valves (a high/low pressure interface).

1997-03, Enforcement Action 97-191, SL III (\$55,000 civil penalty)
Criterion XVI (SL III)

Failure to promptly identify and take corrective actions. Maintenance worker drilled into an electrical junction box, causing fire dampers in two safety-related electrical distribution rooms to actuate. Some dampers did not drop and other became physically restrained and only partially dropped. Condition went unaddressed by plant personnel for two days until questioned by NRC.

1997-08, Enforcement Action 97-367, SL III (\$110,000 civil penalty)
TS 6.8.1 (SL III), Criterion XVI (SL III), TS 3.1.A.4.a (SL III), TS 4.18.c (SL III), TS 4.2.1.(SL IV) - 5 violations

1) operation of the plant for 2.5 days outside technical specifications pressure and temperature curves with the OPS inoperable. Violation of TS 6.8.1.
2) Failed to consider ambient temperature condition on the pressurizer code safety valve set point. Violation of TS 4.2.1 Untimely and ineffective corrective actions. Inadequate 50.59 safety evaluation for a plant mod to remove the pressurizer block house roof. Inoperability of the code safety valves as prescribed by the technical specifications. Numerous opportunities existed for the staff to identify this issue.
3) Ingestion of hose in 21 recirculation pump. Poor engineering resolution to degraded pump performance that preceded the identification of the hose in the 1997 refueling outage. Indications are 21 recirculation pump inoperable since 1995. Inadequate corrective actions.

1997-13, Enforcement Action 97-576, SL III (\$55,000 civil penalty)
Criterion XVI (SL III)

Failure to take prompt and appropriate corrective actions prior to voluntary shutdown in October 1997 to address the recurring DB-50 breaker failures to close on demand.

1997-15, Enforcement Action 98-028, SL IV
Criterion XVI (SL IV), TS 6.8.1 (SL IV) - 2 violations

1) ConEd's failure to address degraded conditions in a timely manner on the post accident containment venting system (PACVS) and the hydrogen recombiner system.
2) An inadequate procedure for operation of the PACVS.

Office of Investigations- January 22, 1998, Enforcement Action 98–056, SL III
50.9 (SL III) - 2 Violations

1) On August 8, 1997, the emergency battery lights in the PAB were not tested per procedure. However, records were created that indicated the lights were tested. Technicians were not in room for long enough period to adequately test lights.

2) On August 8, 1997, surveillance test of EDG auxiliaries require double verification. Double verification of compressor was not performed. Records were created that indicate second verification was performed. Technician was not in the EDG building to be able to perform verification.

1998-02, Enforcement Action 98-192, SL III ($55,000 civil penalty)
Criterion XI (SL III)

A significant number of technical surveillance testing discrepancies were identified through ConEd and NRC reviews. Failed to assure that all testing required to demonstrate that systems and components will perform satisfactorily in service, as specified in technical specifications, was incorporated into surveillance test procedures.

1999-014, Enforcement Action 99-319, SL II ($88,000 civil penalty)
Criterion III (2 violations), Criterion V, Criterion XVI (SL II)

1) a. Design basis not correctly translated into specifications and procedures for mod to the 480 vital bus degraded voltage relays. Therefore, relays could not perform design basis function and correctly reset. Contributing to August 31, 1999 transfer of 480V bus from offsite power supply to the RDGs.
b. Requirement for auto operation of the Station Aux Transformer Load Tap Changer were not translated into procedures. As a result form September 9, 1998 to August 31, 1999, the 138kV offsite power system was unable to perform its function. Violated Technical specification 3.7. B.3.
2) Procedure did not adequately ensure proper calibration of DB-75 breaker trip units for the EDGs. Result EDG was inoperable from May 27, 1999 through August 31, 1999.
3) Condition adverse to quality with channel 4 of the reactor protection system (RPS) OTDT circuitry between January 1999 and August 31, 1999, resulting in a plant trip during maintenance on channel 3.

2001-010, Enforcement Action 00-179, Red Finding
Criterion XVI (Red)

A PWSCC defect was identified, signifying the potential for other similar cracks in low-row tubes. ConEd did not adequately evaluate the susceptibility for low-row tubes to PWSCC and the extent of degradation.
ConEd did not adequately evaluate the potential for hour-glassing based on the indications of the low-row tube denting. The increased stresses caused by the hour-glassing are a prime precursor for PWSCC.
1997 Steam generator inspection program was not adjusted to compensate for the

adverse effects of increased noise in detecting flaws, particularly when condition that increased the susceptibility to PWSCC existed.

These problems contributed to at least four tubes with PWSCC flaws in their small radius U-bends, being left in service following the 1997 inspection, until one tube failed on February 15, 2000.

EXHIBIT X c1

Report to Congressional Requesters:

May 2004:

Nuclear Regulation:

NRC's Liability Insurance Requirements for Nuclear Power Plants Owned by Limited Liability Companies:

[Hyperlink, http://www.gao.gov/cgi-bin/getrpt?GAO-04-654]:

GAO Highlights:

Highlights of GAO-04-654, a report to congressional requesters

Why GAO Did This Study:

An accident at one the nation's commercial nuclear power plants could result in human health and environmental damages. To ensure that funds would be available to settle liability claims in such cases, the Price-Anderson Act requires licensees for these plants to have primary insurance—currently $300 million per site. The act also requires secondary coverage in the form of retrospective premiums to be contributed by all licensees to cover claims that exceed primary insurance. If these premiums are needed, each licensee's payments are limited to $10 million per year and $95.8 million in total for each of its plants. In recent years, limited liability companies have increasingly become licensees of nuclear power plants, raising concerns about whether these companies—by shielding their parent corporations' assets—will have the financial resources to pay their retrospective premiums.

GAO was asked to determine (1) the extent to which limited liability companies are the licensees for U.S. commercial nuclear power plants, (2) the Nuclear Regulatory Commission's (NRC) requirements and procedures for ensuring that licensees of nuclear power plants comply with the Price-Anderson Act's liability requirements, and (3) whether and how these procedures differ for licensees that are limited liability companies.

What GAO Found:

Of the 103 operating nuclear power plants, 31 are owned by 11 limited liability companies. Three energy corporations—Exelon, Entergy, and the Constellation Energy Group—are the parent companies for eight of these limited liability companies. These 8 subsidiaries are the licensees or co-licensees for 27 of the 31 plants.

NRC requires all licensees for nuclear power plants to show proof that they have the primary and secondary insurance coverage mandated by the Price-Anderson Act. Licensees obtain their primary insurance through American Nuclear Insurers. Licensees also sign an agreement with NRC to keep the insurance in effect. American Nuclear Insurers also has a contractual agreement with each of the licensees to collect the retrospective premiums if these payments become necessary. A certified copy of this agreement, which is called a bond for payment of retrospective premiums, is provided to NRC as proof of secondary insurance. It obligates the licensee to pay the retrospective premiums to American Nuclear Insurers.

NRC does not treat limited liability companies differently than other licensees with respect to the Price-Anderson Act's insurance requirements. Like other licensees, limited liability companies must show proof of both primary and secondary insurance coverage. American Nuclear Insurers also requires limited liability companies to provide a letter of guarantee from their parent or other affiliated companies with sufficient assets to pay the retrospective premiums. These letters state that the parent or affiliated companies are responsible for paying the retrospective premiums if the limited liability company does not. American Nuclear Insurers informs NRC it has received these letters. In light of the increasing number of plants owned by limited liability companies, NRC is studying its existing regulations and expects to report on its findings by the end of summer 2004.

In commenting on a draft of this report, NRC stated that it accurately reflects the present insurance system for nuclear power plants.

www.gao.gov/cgi-bin/getrpt?GAO-04-654.

To view the full product, including the scope and methodology, click on the link above. For more information, contact Jim Wells at 202-512-3841.

[End of section]

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Appendixes:

Appendix I: Nuclear Power Plant Ownership:

Appendix II: Comments from the Nuclear Regulatory Commission:

Table:

Table 1: Limited Liability Companies Licensed to Operate Nuclear Power Plants and Their Parent Companies:

Letter May 28, 2004:

Congressional Requesters:

An accident at one of the nation's 103[Footnote 1] operating commercial nuclear power plants could result in human health and environmental damages. The Price-Anderson Act was enacted in 1957 to ensure that funds would be available for at least a portion of the damages suffered by the public in the event of an incident at a U.S. nuclear power plant. The act requires each licensee of a nuclear plant to have primary insurance coverage equal to the maximum amount of liability insurance available from private sources--currently $300 million--to settle any such claims against it. In the event of an accident at any plant where liability claims exceed the $300 million primary insurance coverage, the act also requires licensees for all plants to pay retrospective premiums (also referred to as secondary insurance). Under current U.S. Nuclear Regulatory Commission (NRC) regulations, these payments could amount to a maximum of $95.8 million for each of a licensee's plants per incident. If claims for an incident exceed this approximately $10 billion currently available in primary insurance and retrospective premiums, NRC may request additional funds from the Congress. To operate a nuclear power plant, the owner must obtain a license from NRC and meet its regulatory requirements, including those for liability insurance established under the Price-Anderson Act.

A major aspect of the deregulation or restructuring of the U.S. electricity industry in the 1990s was the separation of electricity generation from transmission and distribution. Utilities could create separate entities or subsidiaries to operate their generation facilities, including nuclear power plants, or could sell them off to other companies. Energy holding companies bought some of the generation facilities, sometimes placing them under subsidiaries. The limited liability company also emerged in the 1990s as a new type of company structure in the United States. These companies have characteristics of both a partnership and a corporation. Like a partnership, the profits are passed through and taxable to the owners, known as members; like a corporation, it is a separate and distinct legal entity and its owners are insulated from personal liability for its debts and liabilities.

You asked us to determine (1) the extent to which limited liability companies are the licensees for U.S. commercial nuclear power plants, (2) NRC's requirements and procedures for ensuring that licensees of nuclear power plants comply with the Price-Anderson Act's liability requirements, and (3) whether and how these procedures differ for licensees that are limited liability companies. To respond to your request, we reviewed applicable sections of the Price-Anderson Act and NRC's implementing regulations and written procedures. We also held discussions with and obtained information from responsible NRC officials and representatives of American Nuclear Insurers, which is a joint underwriting association of 50 insurance companies that provides insurance coverage to the nuclear power plants. These are property/casualty insurance companies licensed to do business in at least one of the states or territories of the United States. We performed our work between April 2003 and April 2004 in accordance with generally accepted government auditing standards.

Results in Brief:

Thirty-one of the 103 operating commercial nuclear power plants nationwide are licensed to limited liability companies. Four of the 31 plants are licensed jointly to two limited liability companies. A total of 11 limited liability companies are licensed to own nuclear power plants. One--the Exelon Generation Company, LLC--is the licensee for 12 plants and co-licensee for 4 plants. The 10 other limited liability companies are the licensees or co-licensees for one to five plants. Three energy corporations--Exelon, Entergy, and the Constellation Energy Group--are the parent companies for eight of the limited liability companies. These eight subsidiaries are the licensees or co-licensees for 27 of the 31 plants.

NRC's procedures for ensuring that licensees comply with Price-Anderson Act liability insurance provisions include requirements that licensees provide proof of primary and secondary insurance coverage. NRC requires each licensee to show proof that it has liability insurance that includes the $300 million of primary insurance coverage per site required by the Price-Anderson Act. NRC and the licensee also sign an indemnity agreement that requires the licensee to maintain an insurance policy in this amount. This agreement is in effect as long as the owner is licensed to operate the plant. NRC relies on American Nuclear Insurers--the joint underwriting association that provides insurance for U.S. nuclear power plants--to send NRC the annual endorsements documenting proof of insurance after the licensees have paid their annual premiums. In addition to the primary insurance coverage, licensees must also show proof of secondary insurance to NRC. This secondary insurance is in the form of retrospective premiums that, in the event of a nuclear incident causing damages exceeding $300 million, would be collected from each nuclear power plant licensee at a rate of up to $10 million per year and up to a maximum of $95.8 million per incident for each nuclear power plant. Typically, each licensee signs a bond for payment of retrospective premiums as proof of the secondary insurance and furnishes NRC with a certified copy. This bond is a contractual agreement between the licensee and American Nuclear Insurers that obligates the licensee to pay American Nuclear Insurers the retrospective premiums. In the event that claims exhaust primary coverage, American Nuclear Insurers would collect the retrospective premiums. If a licensee did not pay its share of these retrospective premiums, American Nuclear Insurers would, under its agreement with the licensees, pay up to $30 million of the premiums in 1 year and attempt to collect this amount later from the licensees.

NRC does not treat limited liability companies differently than other licensees of nuclear power plants with respect to Price-Anderson Act liability requirements. All licensees follow the same regulations and procedures regardless of whether they are limited liability companies. Like other licensees, limited liability companies are required to show that they are maintaining $300 million in primary insurance coverage, and they provide NRC a copy of the bond for payment of retrospective premiums. While NRC does not conduct in-depth financial reviews specifically to determine licensees' ability to pay retrospective premiums, when a licensee applies for a license or when the license is transferred, NRC reviews the licensee's financial ability to safely operate the plant and to contribute decommissioning funds for the future retirement of the plant. According to NRC officials, if licensees have the financial resources to cover these two expenses, they are likely to be capable of paying their retrospective premiums. American Nuclear Insurers goes further than NRC and requires limited liability companies to provide a letter of guarantee from their parent or other affiliated companies with sufficient assets to cover the retrospective premiums. These letters state that the parent or an affiliated company is responsible for paying the retrospective premiums if the limited liability company does not. American Nuclear Insurers informs NRC that it has received these letters of guarantee. Recognizing that limited liability companies are becoming more prevalent as owners of nuclear power plants, NRC is examining whether it needs to revise any of its regulations and procedures for these companies. NRC estimates the study will be completed by the end of summer 2004.

In commenting on a draft of this report, NRC stated that it accurately reflects the present insurance system for nuclear power plants.

Background:

The Atomic Energy Act of 1954 authorized a comprehensive regulatory program to permit private industry to develop and apply atomic energy for peaceful uses, such as generating electricity from privately owned nuclear power plants. Soon thereafter, government and industry experts identified a major impediment to accomplishing the act's objective: the potential for payment of damages resulting from a nuclear accident and the lack of adequate available insurance. Unwilling to risk huge financial liability, private companies viewed even the remote specter of a serious accident as a roadblock to their participating in the development and use of nuclear power.[Footnote 2] In addition, congressional concern developed over ensuring adequate financial protection to the public because the public had no assurance that it would receive compensation for personal injury or property damages from the liable party in event of a serious accident. Faced with these concerns, the Congress enacted the Price-Anderson Act in September 1957. The Price-Anderson Act has two underlying objectives: (1) to establish a mechanism for compensating the public for personal injury or property damage in the event of a nuclear accident and (2) to encourage the development of nuclear power.

To provide financial protection, the Price-Anderson Act requires commercial nuclear reactors to be insured to the maximum level of primary insurance available from private insurers. To implement this provision, NRC periodically revises its regulations to require licensees of nuclear reactors to increase their coverage level as the private insurance market increases the maximum level of primary insurance that it is willing to offer. For example, in January 2003, NRC increased the required coverage from $200 million to the current $300 million, when American Nuclear Insurers informed NRC that $300 million per site in coverage was now available in its insurance pool.

In 1975, the Price-Anderson Act was amended to require licensees to pay a pro-rated share of the damages in excess of the primary insurance amount. Under this amendment, each licensee would pay up to $5 million in retrospective premiums per facility it owned per incident if a nuclear accident resulted in damages exceeding the amount of primary insurance coverage. In 1988, the act was further amended to increase the maximum retrospective premium to $63 million per reactor per incident to be adjusted by NRC for inflation. The amendment also limited the maximum annual retrospective premium per reactor to $10 million. Under the act, NRC is to adjust the maximum amount of retrospective premiums every 5 years using the aggregate change in the Consumer Price Index for urban consumers. In August 2003, NRC set the current maximum retrospective payment at $95.8 million per reactor per incident. With 103 operating nuclear power plants, this secondary insurance pool would total about $10 billion.[Footnote 3]

The Price-Anderson Act also provides a process to deal with incidents in which the damages exceed the primary and secondary insurance coverage. Under the act, NRC shall survey the causes and extent of the damage and submit a report on the results to, among others, the Congress and the courts. The courts must determine whether public liability exceeds the liability limits available in the primary insurance and secondary retrospective premiums. Then the President would submit to the Congress an estimate of the financial extent of damages, recommendations for additional sources of funds, and one or more compensation plans for full and prompt compensation for all valid claims. In addition, NRC can request the Congress to appropriate funds. The most serious incident at a U.S. nuclear power plant took place in 1979 at the Three Mile Island Nuclear Station in Pennsylvania. That incident has resulted in $70 million in liability claims.

NRC's regulatory activities include licensing nuclear reactors and overseeing their safe operation. Licensees must meet NRC regulations to obtain and retain their license to operate a nuclear facility. NRC carries out reviews of financial qualifications of reactor licensees when they apply for a license or if the license is transferred, including requiring applicants to demonstrate that they possess or have reasonable assurance of obtaining funds necessary to cover estimated operating costs for the period of the license. NRC does not systematically review its licensees' financial qualifications once it has issued the license unless it has reason to believe this is necessary. In addition, NRC performs inspections to verify that a licensee's activities are properly conducted to ensure safe operations in accordance with NRC's regulations. NRC can issue sanctions to licensees who violate its regulations. These sanctions include notices of violation; civil penalties of up to $100,000 per violation per day; and orders that may modify, suspend, or revoke a license.

Limited Liability Companies Are Licensees for 31 of the 103 Operating Commercial Nuclear Power Plants in the United States:

Thirty-one commercial nuclear power plants nationwide are licensed to limited liability companies. In total, 11 limited liability companies are licensed to own nuclear power plants. Three energy corporations--Exelon, Entergy, and the Constellation Energy Group--are the parent companies for 8 of these limited liability companies. These eight subsidiaries are licensed or co-licensed to operate 27 of the 31 plants. The two subsidiaries of the Exelon Corporation are the licensees for 15 plants and the co-licensees for 4 others. Constellation Energy Group, Inc., and Entergy Corporation are the parent companies of limited liability companies that are licensees for four nuclear power plants each. (See table 1.):

Table 1: Limited Liability Companies Licensed to Operate Nuclear Power Plants and Their Parent Companies:

Limited liability company: Exelon Generation Company, LLC;
Parent company: Exelon Corporation;
Number of plants owned or co-owned: 12.

Limited liability company: AmerGen Energy Company, LLC;
Parent company: Exelon Corporation;
Number of plants owned or co-owned: 3.

Limited liability company: Exelon Generation Company, LLC; PSEG Nuclear, LLC;
Parent company: Exelon Corporation; Public Service Enterprise Group, Incorporated;
Number of plants owned or co-owned: 4.

Limited liability company: PSEG Nuclear, LLC;
Parent company: Public Service Enterprise Group, Incorporated;
Number of plants owned or co-owned: 1.

Limited liability company: Calvert Cliffs Nuclear Power Plant, LLC;
Parent company: Constellation Energy Group, Inc.;
Number of plants owned or co-owned: 2.

Limited liability company: Nine Mile Point Nuclear Station, LLC;
Parent company: Constellation Energy Group, Inc.;
Number of plants owned or co-owned: 2.

Limited liability company: Entergy Nuclear Indian Point 2, LLC;

Parent company: Entergy Corporation;
Number of plants owned or co-owned: 1.

Limited liability company: Entergy Nuclear Indian Point 3, LLC;
Parent company: Entergy Corporation;
Number of plants owned or co-owned: 1.

Limited liability company: Entergy Nuclear FitzPatrick, LLC;
Parent company: Entergy Corporation;
Number of plants owned or co-owned: 1.

Limited liability company: Entergy Nuclear Vermont Yankee, LLC;
Parent company: Entergy Corporation;
Number of plants owned or co-owned: 1.

Limited liability company: FPL Energy Seabrook, LLC;
Parent company: FPL Group, Inc.;
Number of plants owned or co-owned: 1.

Limited liability company: PPL Susquehanna, LLC;
Parent company: Pennsylvania Power and Light Company;
Number of plants owned or co-owned: 2.

Source: GAO survey of NRC project managers.

[End of table]

Of all the limited liability companies, Exelon Generation Company, LLC, has the largest number of plants. It is the licensee for 12 plants and co-licensee with PSEG Nuclear, LLC, for 4 other plants. For these 4 plants, Exelon Generation owns 43 percent of Salem Nuclear Generating Stations 1 and 2 and 50 percent of Peach Bottom Atomic Power Stations 2 and 3. (App. I lists all the licensees and their nuclear power plants.):

NRC Has Specific Requirements and Procedures to Ensure That All Licensees Comply with the Price-Anderson Act's Liability Provisions:

NRC requires licensees of nuclear power plants to comply with the Price-Anderson Act's liability insurance provisions by maintaining the necessary primary and secondary insurance coverage. First, NRC ensures that licensees comply with the primary insurance coverage requirement by requiring them to submit proof of coverage in the amount of $300 million. Second, NRC ensures compliance with the requirement for secondary coverage by accepting the certified copy of the licensee's bond for payment of retrospective premiums.

All the nuclear power plant licensees purchase their primary insurance from American Nuclear Insurers. American Nuclear Insurers sends NRC annual endorsements documenting proof of primary insurance after the licensees have paid their annual premiums. NRC and each licensee also sign an indemnity agreement, stating that the licensee will maintain an insurance policy in the required amount. This agreement, which is in effect as long as the owner is licensed to operate the plant, guarantees reimbursement of liability claims against the licensee in the event of a nuclear incident through the liability insurance. The agency can suspend or revoke the license if a licensee does not maintain the insurance, but according to an NRC official, no licensee has ever failed to pay its annual primary insurance premium and American Nuclear Insurers would notify NRC if a licensee failed to pay.[Footnote 4]

As proof of their secondary insurance coverage, licensees must provide evidence that they are maintaining a guarantee of payment of retrospective premiums. Under NRC regulations, the licensee must provide NRC with evidence that it maintains one of the following six types of guarantees: (1) surety bond, (2) letter of credit, (3) revolving credit/term loan arrangement, (4) maintenance of escrow deposits of government securities, (5) annual certified financial statement showing either that a cash flow can be generated and would be available for payment of retrospective premiums within 3 months after submission of the statement or a cash reserve or combination of these, or (6) such other type of guarantee as may be approved by the Commission.

Before the late 1990s, the licensees provided financial statements to NRC as evidence of their ability to pay retrospective premiums.[Footnote 5] According to NRC officials, in the late 1990s, Entergy asked NRC to accept the bond for payment of retrospective premiums that it had with American Nuclear Insurers as complying with the sixth option under NRC's regulations: such other type of guarantee as may be approved by the Commission. After reviewing and agreeing to Entergy's request, NRC decided to accept the bond from all the licensees as meeting NRC's requirements. NRC officials told us that they did not document this decision with Commission papers or incorporate it into the regulations because they did not view this as necessary under the regulations.

The bond for payment of retrospective premiums is a contractual agreement between the licensee and American Nuclear Insurers that obligates the licensee to pay American Nuclear Insurers the retrospective premiums. Each licensee signs this bond and furnishes NRC with a certified copy. In the event that claims exhaust primary coverage, American Nuclear Insurers would collect the retrospective premiums. If a licensee were not to pay its share of these retrospective premiums, American Nuclear Insurers would, under its agreement with the licensees, pay for up to three defaults or up to $30 million in 1 year of the premiums and attempt to collect this amount later from the defaulting licensees. According to an American Nuclear Insurers official, any additional defaults would reduce the amount available for retrospective payments. An American Nuclear Insurers official told us that his organization believes that the bond for payment of retrospective premiums is legally binding and obligates the licensee to pay the premium. Under NRC regulations, if a licensee fails to pay the assessed deferred premium, NRC reserves the right to pay those premiums on behalf of the licensee and recover the amount of such premiums from the licensee.

NRC Treats Limited Liability Companies the Same as Other Licensees, but the Insurance Industry Has Added Important Requirements for These Companies:

NRC applies the same rules to limited liability companies that it does to other licensees of nuclear power plants with respect to liability requirements under the Price-Anderson Act.

All licensees must meet the same requirements regardless of whether they are limited liability companies. American Nuclear Insurers applies an additional requirement for limited liability companies with respect to secondary insurance coverage in order to ensure that they have sufficient assets to pay retrospective premiums. Given the growing number of nuclear power plants licensed to limited liability companies, NRC is examining the need to revise its procedures and regulations for such companies.

NRC requires all licensees of nuclear power plants to follow the same regulations and procedures. Limited liability companies, like other licensees, are required to show that they are maintaining the $300 million in primary insurance coverage and provide NRC a copy of the bond for payment of retrospective premiums or other approved evidence of guarantee of retrospective premium payments. According to NRC officials, all its licensees, including those that are limited liability companies, have sufficient assets to cover the retrospective premiums. While NRC does not conduct in-depth financial reviews specifically to determine licensees' ability to pay retrospective premiums, it reviews the licensees' financial ability to safely operate their plants and to contribute decommissioning funds for the future retirement of the plants. According to NRC officials, if licensees have the financial resources to cover these two larger expenses, they are likely to be capable of paying their retrospective premiums.

American Nuclear Insurers goes further than NRC and requires licensees that are limited liability companies to provide a letter of guarantee from their parent or other affiliated companies with sufficient assets to cover the retrospective premiums. An American Nuclear Insurers official stated that American Nuclear Insurers obtains these letters as a matter of good business practice. These letters state that the parent or an affiliated company is responsible for paying the retrospective premiums if the limited liability company does not. If the parent company or other affiliated company of a limited liability company does not provide a letter of guarantee, American Nuclear Insurers could refuse to issue the bond for payment of retrospective premiums and the company would have to have another means to show NRC proof of secondary insurance. American Nuclear Insurers informs NRC that it has received these letters of guarantee. The official also told us that American Nuclear Insurers believes that the letters from the parent companies or other affiliated companies of the limited liability company licensed by NRC are valid and legally enforceable contracts.

NRC officials told us that they were not aware of any problems caused by limited liability companies owning nuclear power plants and that NRC currently does not regard limited liability companies' ownership of nuclear power plants as a concern. However, because these companies are becoming more prevalent as owners of nuclear power plants, NRC is examining whether it needs to revise any of its regulations or procedures for these licensees. NRC estimates that it will complete its study by the end of summer 2004.

Agency Comments:

We provided a draft of this report to NRC for review and comment. In its written comments (see app. II), NRC stated that it believes the report accurately reflects the present insurance system for nuclear power plants. NRC said that we correctly conclude that the agency does not treat limited liability companies differently than other licensees with respect to Price-Anderson's insurance requirements. NRC also stated that we are correct in noting that it is not aware of any problems caused by limited liability companies owning nuclear power plants and that NRC currently does not regard limited liability companies' ownership of nuclear power plants as a concern. In addition, NRC commented that we agree with the agency's conclusion that all its reactor licensees have sufficient assets that they are likely to be able to pay the retrospective premiums. With respect to this last comment, the report does not take a position on the licensees' ability to pay the retrospective premiums. We did not evaluate the sufficiency of the individual licensees' assets to make these payments. Instead, we reviewed NRC's and the American Nuclear Insurers' requirements and procedures for retrospective premiums.

Scope and Methodology:

We performed our review at NRC headquarters in Washington, D.C. We reviewed statutes, regulations, and appropriate guidance as well as interviewed agency officials to determine the relevant statutory framework of the Price-Anderson Act. To determine the number of nuclear power plant licensees that are limited liability companies, we surveyed, through electronic mail, all the NRC project managers responsible for maintaining nuclear power plant licenses. We asked them to provide data on the licensees, including the licensee's name and whether it was a limited liability company. If it was a limited liability company, we asked when the license was transferred to the limited liability company and who is the parent company of the limited liability company. We received responses for all 103 nuclear power plants currently licensed to operate. We analyzed the results of the survey responses. We verified the reliability of the data from a random sample of project managers by requesting copies of the power plant licenses and then comparing the power plant licenses to the data provided by the project managers. The data agreed in all cases. We concluded that the data were reliable enough for the purposes of this report.

To determine NRC's requirements for ensuring that licensees of nuclear power plants comply with the Price-Anderson Act's liability requirements, we reviewed relevant statutes and NRC regulations and interviewed NRC officials responsible for ensuring that licensees have primary and secondary insurance coverage. We also spoke with American Nuclear Insurers officials responsible for issuing the insurance coverage to nuclear power plant licensees, and we reviewed relevant documents associated with the insurance. To determine whether and how these procedures differ for licensees that are limited liability companies, we reviewed relevant documents, including NRC regulations, and interviewed NRC officials responsible for ensuring licensee compliance with Price-Anderson Act requirements.

As agreed with your offices, unless you publicly announce its contents earlier, we plan no further distribution of this report until 7 days from the date of this letter. We will then send copies to interested congressional committees; the Commissioners, Nuclear Regulatory Commission; the Director, Office of Management and Budget; and other interested parties. We will make copies available to others on request. In addition, the report will be available at no charge on GAO's Web site at [Hyperlink, http://www.gao.gov].

If you or your staff have any questions about this report, I can be reached at (202) 512-3841. Major contributors to this report include Ray Smith, Ilene Pollack, and Amy Webbink. John Delicath and Judy Pagano also contributed to this report.

Signed by:

Jim Wells,
Director, Natural Resources and Environment:

List of Congressional Requesters:

The Honorable Hillary Rodham Clinton:
The Honorable James M. Jeffords:
The Honorable Harry Reid:
United States Senate:

[End of section]

Appendixes:

Appendix I: Nuclear Power Plant Ownership:

1;
Plant: Arkansas Nuclear One 1;
Licensed to own: Entergy Arkansas, Inc.;
LLC: No.

2;
Plant: Arkansas Nuclear One 2;
Licensed to own: Entergy Arkansas, Inc.;
LLC: No.

3;
Plant: Arnold (Duane) Energy Center;

Licensed to own: Interstate Power and Light;
LLC: No;
Licensed to own: Central Iowa Power Cooperative;
LLC: No;
Licensed to own: Corn Belt Power Cooperative;
LLC: No.

4;
Plant: Beaver Valley Power Station 1;
Licensed to own: Pennsylvania Power Company;
LLC: No.
Licensed to own: Ohio Edison Company;
LLC: No;
Licensed to own: FirstEnergy Nuclear Operating Company;
LLC: No.

5;
Plant: Beaver Valley Power Station 2;
Licensed to own: Pennsylvania Power Company;
LLC: No;
Licensed to own: Ohio Edison Company;
LLC: No;
Licensed to own: Cleveland Electric Illuminating Company;
LLC: No;
Licensed to own: Toledo Edison Company;
LLC: No;
Licensed to own: FirstEnergy Nuclear Operating Company;
LLC: No.

6;
Plant: Braidwood Station 1;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation.

7;
Plant: Braidwood Station 2;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation.

8;
Plant: Browns Ferry Nuclear Power Station 1;
Licensed to own: Tennessee Valley Authority;
LLC: No.

9;
Plant: Browns Ferry Nuclear Power Station 2;
Licensed to own: Tennessee Valley Authority;
LLC: No.

10;
Plant: Browns Ferry Nuclear Power Station 3;
Licensed to own: Tennessee Valley Authority;
LLC: No.

11;
Plant: Brunswick Steam Electric Plant 1;
Licensed to own: Carolina Power & Light Co.;
LLC: No;
Licensed to own: North Carolina Eastern Municipal Power Agency;
LLC: No.

12;
Plant: Brunswick Steam Electric Plant 2;

Licensed to own: Carolina Power & Light Co.;
LLC: No;
Licensed to own: North Carolina Eastern Municipal Power Agency;
LLC: No.

13;
Plant: Byron Station 1;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation.

14;
Plant: Byron Station 2;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation.

15;
Plant: Callaway Plant;
Licensed to own: Union Electric Company;
LLC: No.

16;
Plant: Calvert Cliffs Nuclear Power Plant 1;
Licensed to own: Calvert Cliffs Nuclear Power Plant, LLC;
LLC: Yes;
License transfer date: 6/19/2001;
LLC parent company: Constellation Energy Group, Inc..

17;
Plant: Calvert Cliffs Nuclear Power Plant 2;
Licensed to own: Calvert Cliffs Nuclear Power Plant, LLC;
LLC: Yes;
License transfer date: 6/19/2001;
LLC parent company: Constellation Energy Group, Inc..

18;
Plant: Catawba Nuclear Station 1;
Licensed to own: North Carolina Electric Membership Corp.;
LLC: No;
Licensed to own: Saluda River Electric Cooperative, Inc.;
LLC: No;
Licensed to own: Duke Energy Corporation;
LLC: No.

19;
Plant: Catawba Nuclear Station 2;
Licensed to own: North Carolina Municipal Power Agency No. 1;
LLC: No;
Licensed to own: Piedmont Municipal Power Agency;
LLC: No.

20;
Plant: Clinton Power Station;
Licensed to own: AmerGen Energy Company, LLC;
LLC: Yes;
License transfer date: 11/24/1999;
LLC parent company: Exelon Corporation.

21;
Plant: Columbia Generation Station;
Licensed to own: Energy Northwest;
LLC: No.

22;

Plant: Comanche Peak Steam Electric Station 1;
Licensed to own: TXU Generation Company LP;
LLC: No.

23;
Plant: Comanche Peak Steam Electric Station 2;
Licensed to own: TXU Generation Company LP;
LLC: No.

24;
Plant: Cook (Donald C.) Nuclear Power Plant 1;
Licensed to own: Indiana Michigan Power Company;
LLC: No.

25;
Plant: Cook (Donald C.) Nuclear Power Plant 2;
Licensed to own: Indiana Michigan Power Company;
LLC: No.

26;
Plant: Cooper Nuclear Station;
Licensed to own: Nebraska Public Power District;
LLC: No.

27;
Plant: Crystal River Nuclear Plant 3;
Licensed to own: Florida Power Corporation;
LLC: No;
Licensed to own: City of Alachua;
LLC: No;
Licensed to own: City of Bushnell;
LLC: No;
Licensed to own: City of Gainesville;
LLC: No;
Licensed to own: City of Kissimmee;
LLC: No;
Licensed to own: City of Leesburg;
LLC: No;
Licensed to own: City of New Smyrna Beach and Utilities Commission;
LLC: No;
Licensed to own: City of Ocala;
LLC: No;
Licensed to own: Orlando Utilities Commission and City of Orlando;
LLC: No;
Licensed to own: Seminole Electric Cooperative, Inc.;
LLC: No.

28;
Plant: Davis-Besse Nuclear Power Station;
Licensed to own: Cleveland Electric Illumination Company;
LLC: No;
Licensed to own: Toledo Edison Company;
LLC: No.

29;
Plant: Diablo Canyon Nuclear Power Plant 1;
Licensed to own: Pacific Gas and Electric Company;
LLC: No.

30;
Plant: Diablo Canyon Nuclear Power Plant 2;
Licensed to own: Pacific Gas and Electric Company;
LLC: No.

31;
Plant: Dresden Nuclear Power Station 2;
Licensed to own: Exelon Generation Company, LLC;

LLC: Yes;
License transfer date: 8/3/2000;
LLC parent company: Exelon Corporation.

32;
Plant: Dresden Nuclear Power Station 3;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 8/3/2000;
LLC parent company: Exelon Corporation.

33;
Plant: Farley (Joseph M.) Nuclear Plant 1;
Licensed to own: Alabama Power Company;
LLC: No.

34;
Plant: Farley (Joseph M.) Nuclear Plant 2;
Licensed to own: Alabama Power Company;
LLC: No.

35;
Plant: Fermi (Enrico) Atomic Power Plant 2;
Licensed to own: Detroit Edison Company;
LLC: No.

36;
Plant: FitzPatrick (James A.) Nuclear Power Plant;
Licensed to own: Entergy Nuclear FitzPatrick, LLC;
LLC: Yes;
License transfer date: 11/ 21/2000;
LLC parent company: Entergy Corporation.

37;
Plant: Fort Calhoun Station;
Licensed to own: Omaha Public Power District;
LLC: No.

38;
Plant: Ginna (Robert E.) Nuclear Station;
Licensed to own: Rochester Gas and Electric Corporation;
LLC: No.

39;
Plant: Grand Gulf Nuclear Station;
Licensed to own: System Energy Resources, Inc.;
LLC: No;
Licensed to own: South Mississippi Electric Power Assoc.;
LLC: No.

40;
Plant: Harris (Shearon) Nuclear Power Plant;
Licensed to own: Carolina Power & Light Co.;
LLC: No;
Licensed to own: North Carolina Eastern Municipal Power Agency;
LLC: No.

41;
Plant: Hatch (Edwin I.) Nuclear Plant 1;
Licensed to own: Georgia Power Company;
LLC: No;
Licensed to own: Municipal Electric Authority of Georgia;
LLC: No;
Licensed to own: Oglethorpe Power Corporation;
LLC: No;
Licensed to own: City of Dalton, Georgia;
LLC: No.

42;
Plant: Hatch (Edwin I.) Nuclear Plant 2;
Licensed to own: Georgia Power Company;
LLC: No;
Licensed to own: Municipal Electric Authority of Georgia;
LLC: No;
Licensed to own: Oglethorpe Power Corporation;
LLC: No;
Licensed to own: City of Dalton, Georgia;
LLC: No.

43;
Plant: Hope Creek Nuclear Power Station;
Licensed to own: PSEG Nuclear, LLC;
LLC: Yes;
License transfer date: 8/21/2000;
10/18/2001;
LLC parent company: Public Service Enterprise Group, Incorporated.

44;
Plant: Indian Point 2;
Licensed to own: Entergy Nuclear Indian Point 2, LLC;
LLC: Yes;
License transfer date: 9/6/2001;
LLC parent company: Entergy Corporation.

45;
Plant: Indian Point 3;
Licensed to own: Entergy Nuclear Indian Point 3, LLC;
LLC: Yes;
License transfer date: 11/21/2000;
LLC parent company: Entergy Corporation.

46;
Plant: Kewaunee Nuclear Power Plant;
Licensed to own: Wisconsin Public Service Corp.;
LLC: No;
Licensed to own: Wisconsin Power & Light Company;
LLC: No.

47;
Plant: LaSalle County Station 1;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation.

48;
Plant: LaSalle County Station 2;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation.

49;
Plant: Limerick Generating Station 1;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation.

50;
Plant: Limerick Generating Station 2;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;

LLC parent company: Exelon Corporation.

51;
Plant: McGuire (William B.) Nuclear Station 1;
Licensed to own: Duke Energy Corporation;
LLC: No.

52;
Plant: McGuire (William B.) Nuclear Station 2;
Licensed to own: Duke Energy Corporation;
LLC: No.

53;
Plant: Millstone Nuclear Power Station 2;
Licensed to own: Dominion Nuclear Connecticut, Inc.;
LLC: No.

54;
Plant: Millstone Nuclear Power Station 3;
Licensed to own: Dominion Nuclear Connecticut, Inc.;
LLC: No;
Licensed to own: Central Vermont Public Service Corporation;
LLC: No;
Licensed to own: Massachusetts Municipal Wholesale Electric Co.;
LLC: No.

55;
Plant: Monticello Nuclear Generating Plant;
Licensed to own: Northern States Power Company;
LLC: No.

56;
Plant: Nine Mile Point Nuclear Station 1;
Licensed to own: Nine Mile Point Nuclear Station, LLC;
LLC: Yes;
License transfer date: 11/7/ 2001;
LLC parent company: Constellation Energy Group.

57;
Plant: Nine Mile Point Nuclear Station 2;
Licensed to own: Nine Mile Point Nuclear Station, LLC;
LLC: Yes;
License transfer date: 11/7/ 2001;
LLC parent company: Constellation Energy Group;
Licensed to own: Long Island Lighting Company;
LLC: No.

58;
Plant: North Anna Power Station 1;
Licensed to own: Virginia Electric and Power Company;
LLC: No;
Licensed to own: Old Dominion Electric Cooperative;
LLC: No.

59;
Plant: North Anna Power Station 2;
Licensed to own: Virginia Electric and Power Company;
LLC: No;
Licensed to own: Old Dominion Electric Cooperative;
LLC: No.

60;
Plant: Oconee Nuclear Station 1;
Licensed to own: Duke Energy Corporation;
LLC: No.

61;

Plant: Oconee Nuclear Station 2;
Licensed to own: Duke Energy Corporation;
LLC: No.

62;
Plant: Oconee Nuclear Station 3;
Licensed to own: Duke Energy Corporation;
LLC: No.

63;
Plant: Oyster Creek Nuclear Power Plant;
Licensed to own: AmerGen Energy Company, LLC;
LLC: Yes;
License transfer date: 8/8/2000;
LLC parent company: Exelon Corporation.

64;
Plant: Palisades Nuclear Plant;
Licensed to own: Consumers Energy Company;
LLC: No.

65;
Plant: Palo Verde Nuclear Generating Station 1;
Licensed to own: Arizona Public Service Company;
LLC: No;
Licensed to own: Salt River Project Agricultural Improvement and Power District;
LLC: No;
Licensed to own: El Paso Electric Company;
LLC: No;
Licensed to own: Southern California Edison Company;
LLC: No;
Licensed to own: Public Service Company of New Mexico;
LLC: No;
Licensed to own: Los Angeles Dept. of Water and Power;
LLC: No;
Licensed to own: Southern California Public Power Authority;
LLC: No.

66;
Plant: Palo Verde Nuclear Generating Station 2;
Licensed to own: Arizona Public Service Company;
LLC: No;
Licensed to own: Salt River Project Agricultural Improvement and Power District;
LLC: No;
Licensed to own: El Paso Electric Company;
LLC: No;
Licensed to own: Southern California Edison Company;
LLC: No;
Licensed to own: Public Service Company of New Mexico;
LLC: No;
Licensed to own: Los Angeles Dept. of Water and Power;
LLC: No;
Licensed to own: Southern California Public Power Authority;
LLC: No.

67;
Plant: Palo Verde Nuclear Generating Station 3;
Licensed to own: Arizona Public Service Company;
LLC: No;
Licensed to own: Salt River Project Agricultural Improvement and Power District;
LLC: No;
Licensed to own: El Paso Electric Company;
LLC: No;
Licensed to own: Southern California Edison Company;

LLC: No;
Licensed to own: Public Service Company of New Mexico;
LLC: No;
Licensed to own: Los Angeles Dept. of Water and Power;
LLC: No;
Licensed to own: Southern California Public Power Authority;
LLC: No.

68;
Plant: Peach Bottom Atomic Power Station 2;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation;
Licensed to own: PSEG Nuclear, LLC;
LLC: Yes.

LLC parent company: Public Service Enterprise Group, Incorporated.

69;
Plant: Peach Bottom Atomic Power Station 3;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation;
Licensed to own: PSEG Nuclear, LLC;
LLC: Yes.

LLC parent company: Public Service Enterprise Group, Incorporated.

70;
Plant: Perry Nuclear Power Plant;
Licensed to own: Ohio Edison Company;
LLC: No;
Licensed to own: Cleveland Electric Company;
LLC: No;
Licensed to own: Toledo Edison Company;
LLC: No.

71;
Plant: Pilgrim Station;
Licensed to own: Entergy Nuclear Generation Co.;
LLC: No.

72;
Plant: Point Beach Nuclear Plant 1;
Licensed to own: Wisconsin Electric Power Company;
LLC: No.

73;
Plant: Point Beach Nuclear Plant 2;
Licensed to own: Wisconsin Electric Power Company;
LLC: No.

74;
Plant: Praire Island Nuclear Plant 1;
Licensed to own: Northern States Power Company;
LLC: No.

75;
Plant: Praire Island Nuclear Plant 2;
Licensed to own: Northern States Power Company;
LLC: No.

76;
Plant: Quad Cities Station 1;
Licensed to own: Exelon Generation Company, LLC;

LLC: Yes;
License transfer date: 8/3/2000;
LLC parent company: Exelon Corporation;
Licensed to own: 77: MidAmerican Energy Company;
LLC: 77: No;
License transfer date: 77: [Empty];
LLC parent company: 77: [Empty].

77;
Plant: Quad Cities Station 2;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 8/3/2000;
LLC parent company: Exelon Corporation;
Licensed to own: MidAmerican Energy Company;
LLC: No.

78;
Plant: River Bend Station;
Licensed to own: Entergy Gulf States, Inc.;
LLC: No.

79;
Plant: Robinson (H. B.) Plant 2;
Licensed to own: Carolina Power & Light Co.;
LLC: No.

80;
Plant: Salem Nuclear Generating Station 1;
Licensed to own: PSEG Nuclear, LLC;
LLC: Yes;
License transfer date: 8/21/2000;
LLC parent company: Public Service Enterprise Group, Incorporated;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation.

81;
Plant: Salem Nuclear Generating Station 2;
Licensed to own: PSEG Nuclear, LLC;
LLC: Yes;
License transfer date: 8/21/2000;
LLC parent company: Public Service Enterprise Group, Incorporated;
Licensed to own: Exelon Generation Company, LLC;
LLC: Yes;
License transfer date: 1/12/2001;
LLC parent company: Exelon Corporation.

82;
Plant: San Onofre Nuclear Generating Station 2;
Licensed to own: Southern California Edison Company;
LLC: No.

83;
Plant: San Onofre Nuclear Generating Station 3;
Licensed to own: Southern California Edison Company;
LLC: No.

84;
Plant: Seabrook Nuclear Power Station;
Licensed to own: FPL Energy Seabrook, LLC;
LLC: Yes;
License transfer date: 11/1/2002;
LLC parent company: FPL Group, Inc.;
Licensed to own: Massachusetts Municipal Wholesale Electric Co.;
LLC: No;

Licensed to own: Tauton Municipal Lighting Plant;
LLC: No;
Licensed to own: Hudson Light & Power Department;
LLC: No.

85;
Plant: Sequoya Nuclear Plant 1;
Licensed to own: Tennessee Valley Authority;
LLC: No.

86;
Plant: Sequoya Nuclear Plant 2;
Licensed to own: Tennessee Valley Authority;
LLC: No.

87;
Plant: South Texas Project 1;
Licensed to own: Texas Genco, LP;
LLC: No;
Licensed to own: City Public Service Board of San Antonio;
LLC: No;
Licensed to own: Central Power & Light Company;
LLC: No;
Licensed to own: City of Austin, Texas;
LLC: No.

88;
Plant: South Texas Project 2;
Licensed to own: Texas Genco, LP;
LLC: No;
Licensed to own: City Public Service Board of San Antonio;
LLC: No;
Licensed to own: Central Power & Light Company;
LLC: No;
Licensed to own: City of Austin, Texas;
LLC: No.

89;
Plant: St. Lucie Plant 1;
Licensed to own: Florida Power and Light Company;
LLC: No.

90;
Plant: St. Lucie Plant 2;
Licensed to own: Florida Power and Light Company;
LLC: No;
Licensed to own: Florida Municipal Power Agency;
LLC: No;
Licensed to own: Orlando Utilities Commission;
LLC: No.

91;
Plant: Summer (Virgil C.) Nuclear Station;
Licensed to own: South Carolina Electric & Gas Company;
LLC: No;
Licensed to own: South Carolina Public Service Authority;
LLC: No.

92;
Plant: Surry Power Station 1;
Licensed to own: Virginia Electric and Power Company;
LLC: No.

93;
Plant: Surry Power Station 2;
Licensed to own: Virginia Electric and Power Company;
LLC: No.

94;
Plant: Susquehanna Steam Electric Station 1;
Licensed to own: PPL Susquehanna, LLC;
LLC: Yes;
License transfer date: 6/1/2000;
LLC parent company: Pennsylvania Power and Light Company.

95;
Plant: Susquehanna Steam Electric Station 2;
Licensed to own: PPL Susquehanna, LLC;
LLC: Yes;
License transfer date: 6/1/2000;
LLC parent company: Pennsylvania Power and Light Company.

96;
Plant: Three Mile Island Nuclear Station 1;
Licensed to own: AmerGen Energy Company, LLC;
LLC: Yes;
License transfer date: 12/20/ 1999;
LLC parent company: Exelon Corporation.

97;
Plant: Turkey Point Station 3;
Licensed to own: Florida Power and Light Company;
LLC: No.

98;
Plant: Turkey Point Station 4;
Licensed to own: Florida Power and Light Company;
LLC: No.

99;
Plant: Vermont Yankee Nuclear Power Station;
Licensed to own: Entergy Nuclear Vermont Yankee, LLC;
LLC: Yes;
License transfer date: 7/1/2002;
LLC parent company: Entergy Corporation;
Licensed to own: Entergy Nuclear Operations, Inc.;
LLC: No.

100;
Plant: Vogtle (Alvin W.) Nuclear Plant 1;
Licensed to own: Georgia Power Company;
LLC: No;
Licensed to own: Municipal Electric Authority of Georgia;
LLC: No;
Licensed to own: Oglethorpe Power Corporation;
LLC: No;
Licensed to own: City of Dalton, Georgia;
LLC: No.

101;
Plant: Vogtle (Alvin W.) Nuclear Plant 2;
Licensed to own: Georgia Power Company;
LLC: No;
Licensed to own: Municipal Electric Authority of Georgia;
LLC: No;
Licensed to own: Oglethorpe Power Corporation;
LLC: No;
Licensed to own: City of Dalton, Georgia;
LLC: No.

102;
Plant: Waterford Generating Station 3;
Licensed to own: Entergy Operations, Inc.;
LLC: No.

103;
Plant: Watts Bar Nuclear Plant 1;
Licensed to own: Tennessee Valley Authority;
LLC: No.

104;
Plant: Wolf Creek Generating Station;
Licensed to own: Kansas Gas & Electric Company;
LLC: No;
Licensed to own: Kansas City Power & Light Company;
LLC: No;
Licensed to own: Kansas Electric Power Cooperative, Inc.;
LLC: No.

Source: GAO survey of NRC Project Managers.

[End of table]

[End of section]

Appendix II: Comments from the Nuclear Regulatory Commission:

UNITED STATES NUCLEAR REGULATORY COMMISSION:
WASHINGTON, D.C. 20555-0001:

April 29, 2004:

Mr. James E. Wells:

Director, Natural Resources and Environment:
United States General Accounting Office:
441 G Street, N.W.
Washington, DC 20548:

Dear Mr. Wells:

I would like to thank you for the opportunity to review and submit comments on the May 2004 draft of the General Accounting Office's (GAO) report entitled "Nuclear Regulation-NBC's Liability Insurance Requirements for Nuclear Power Plants Owned by Limited Liability Companies." The U.S. Nuclear Regulatory Commission (NRC) appreciates the time and effort that you and your staff have taken to review this topic.

GAO correctly concludes that NRC does not treat limited liability companies differently than other licensees with respect to the Price-Anderson's insurance requirements. Like other licensees, limited liability companies must show proof of both primary and secondary financial protection. GAO also is correct in noting that NRC is not aware of any problems caused by limited liability companies owning nuclear power plants and that NRC currently does not regard limited liability companies' ownership of nuclear power plants as a concern. Finally, GAO agrees with NBC's conclusion that all its reactor licensees have sufficient assets that they are likely to be able to pay the retrospective premiums. These assets are assured by a number of different methods that are approved by NRC as GAO discusses in its report.

The NRC believes that the GAO report accurately reflects the present insurance system for nuclear power plants. Therefore, we do not have any comments to provide regarding the draft report.

Sincerely,

Signed by:

William D. Travers:
Executive Director for Operations:

cc: Ilene Pollack, GAO:

(360330):

FOOTNOTES

[1] Although 104 commercial nuclear power plants are licensed to operate in the United States, 1 plant, Browns Ferry Unit 1, was shut down in 1985 and remains idle.

[2] NRC's regulations define a nuclear incident as any occurrence that causes bodily injury, sickness, disease, or death or loss of or damage to property or for loss of the use of property arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of the source, special nuclear or byproduct material.

[3] NRC regulations also require licensees to maintain $1 billion in on-site property damage insurance to provide funds to deal with cleanup of the reactor site after an accident.

[4] The average annual premium for a single nuclear power plant at a site is about $400,000. The premium for a second or third plant at the same site is discounted because the maximum amount of primary insurance for a multi-plant site is $300 million.

[5] Fifteen licensees continue to provide financial statements to NRC.

U.S. General Accounting Office

441 G Street NW,

Room LM Washington,

D.C. 20548:

To order by Phone:

Voice: (202) 512-6000:

TDD: (202) 512-2537:

Fax: (202) 512-6061:

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Automated answering system: (800) 424-5454 or (202) 512-7470:

Public Affairs:

Jeff Nelligan, managing director, NelliganJ@gao.gov (202) 512-4800 U.S. General Accounting Office, 441 G Street NW, Room 7149 Washington, D.C. 20548:

Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601-1813
Tel 914 272 3500

March 28, 2005
BVY-05-033
NL-05-039
JPN-05-005
ENO Ltr. 2.05.023

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Stations 1, 2 and 3
Docket Nos. 50-3, 50-247 and 50-286
Vermont Yankee Nuclear Power Station
Docket No. 50-271
Pilgrim Nuclear Power Station
Docket No. 50-293
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Status of Decommissioning Funding for
Plants Operated by Entergy Nuclear Operations, Inc.
For Year Ending December 31, 2004 – 10 CFR 50.75(f)(1)

References:

1. NUREG-1307, "Report on Waste Burial Charges," Revision 10, dated October 2002.
2. NRC Regulatory Issue Summary 2001-07, "10 CFR 50.75(f)(1) Reports on the Status of Decommissioning Funds (Due March 31, 2001)."

Dear Sir:

10 CFR 50.75(f)(1) requires each power reactor licensee to report to the NRC by March 31, 1999, and every two years thereafter, on the status of its decommissioning funding for each reactor, or share of a reactor, that it owns. On behalf of Entergy Nuclear Indian Point 2 LLC, Entergy Nuclear Indian Point 3 LLC, Entergy Nuclear Vermont Yankee LLC, Entergy Nuclear Generation Company (Pilgrim Station), and Entergy Nuclear FitzPatrick LLC, Entergy Nuclear Operations, Inc. hereby submits the information requested for power reactors operated by Entergy Nuclear Operations, Inc.

A001

The estimated minimum decommissioning fund values were determined using the NRC's methodology in NUREG-1307 (Reference 1) and does not include activities outside of the scope of decommissioning as defined by 10 CFR 50.2.

Entergy will continue to monitor the status of the funds and will assure that we meet all NRC regulations and implement NRC guidance, as appropriate, to assure adequate funding for the decommissioning when required.

The information provided in Attachment 1 is based on NRC Regulatory Issue Summary 2001-07 (Reference 2).

There are no new commitments made in this letter. If you have any questions, please contact Ms. Charlene Faison at 914-272-3378.

Very truly yours,

Fred R. Dacimo
Acting Senior Vice President and
Chief Operating Officer

Attachment:

1. Status of Decommissioning Funding for Plants Operated by Entergy Nuclear Operations, Inc. (Indian Point 1, Indian Point 2, Indian Point 3, Vermont Yankee, Pilgrim, and FitzPatrick) For Year Ending December 31, 2004 – 10 CFR 50.75(f)(1) - (7 sheets)

2. NRC Minimum Funding Calculation (10 CFR 50.75(c)) for Indian Point 1, Indian Point 2, Indian Point 3, Vermont Yankee, Pilgrim, and FitzPatrick - (6 sheets)

cc: Next page.

cc: all w/attachments

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. P. Milano, Senior Project Manager
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop: O-8-C2
Washington, DC 20555-0001

Resident Inspector's Office
Indian Point 3 Nuclear Power Plant
U. S. Nuclear Regulatory Commission
P. O. Box 337
Buchanan, NY 10511

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U. S. Nuclear Regulatory Commission
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Resident Inspector's Office
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U. S. Nuclear Regulatory Commission
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Lycoming, NY 13093

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U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
Pilgrim Nuclear Power Station
600 Rocky Hill Road
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Plymouth, MA 02360

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U. S. Nuclear Regulatory Commission
Mail Stop O-8B-1
Washington, DC 20555-0001

USNRC Resident Inspector
Vermont Yankee Nuclear Power Station
320 Governor Hunt Road
P. O. Box 157
Vernon, VT 05354

Mr. David O'Brien, Commissioner
Department of Public Service
120 State Street – Drawer 20
Montpelier, VT 05602

Mr. Paul Eddy
NYS Department of Public Service
3 Empire State Plaza
Albany, NY 12223

Attachment 1 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
Status of Decommissioning Funding
For Year Ending December 31, 2004 – 10 CFR 50.75(f)(1)

Plant Name: **Indian Point Nuclear Generating Unit No. 1**

1.	Amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75 (b) and (c).	$ 309.59 million [1][3]
	Decommissioning cost estimate escalated at 3.0% per year to the midpoint of decommissioning (December 2016).	$ 441.40 million
2.	Amount accumulated to the end of the calendar year preceding the date of the report (December 31, 2004).	$ 227.43 million [2]
	Fund balance with 5.0% annual growth to the midpoint of decommissioning (December 2016).	$ 408.43 million
3.	A schedule of the annual amounts remaining to be collected.	None.
4.	Assumptions used in determining rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections.	Escalation rate: 3.0% Rate of earnings: 5.0%
5.	Any contracts upon which the licensee is relying pursuant to 10 CFR 50.75(e)(1)(v).	None.
6.	Modifications occurring to a licensee's current method of providing financial assurance since the last submitted report.	None.
7.	Any material changes to trust agreements.	None.

Attachment 1 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
Status of Decommissioning Funding
For Year Ending December 31, 2004 – 10 CFR 50.75(f)(1)

Plant Name: **Indian Point Nuclear Generating Unit No. 2**

1.	Amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75 (b) and (c).	$ 373.79 million [1]
	Decommissioning cost estimate escalated at 3.0% per year to the midpoint of decommissioning (December 2016).	$ 532.94 million
2.	Amount accumulated to the end of the calendar year preceding the date of the report (December 31, 2004).	$ 272.04 million [2]
	Fund balance with 5.0% annual growth to the midpoint of decommissioning (December 2016).	$ 488.54 million
3.	A schedule of the annual amounts remaining to be collected.	None.
4.	Assumptions used in determining rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections.	Escalation rate: 3.0% Rate of earnings: 5.0%
5.	Any contracts upon which the licensee is relying pursuant to 10 CFR 50.75(e)(1)(v).	None.
6.	Modifications occurring to a licensee's current method of providing financial assurance since the last submitted report.	None.
7.	Any material changes to trust agreements.	None.

Attachment 1 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
Status of Decommissioning Funding
For Year Ending December 31, 2004 – 10 CFR 50.75(f)(1)

Plant Name: **Indian Point Nuclear Generating Unit No. 3**

1.	Amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75 (b) and (c).	\$ 369.06 million [1]
	Decommissioning cost estimate escalated at 3.0% per year to the midpoint of decommissioning (December 2018).	\$ 558.24 million
2.	Amount accumulated to the end of the calendar year preceding the date of the report (December 31, 2004).	\$ 393.00 million
	Fund balance with 5.0% annual growth to the midpoint of decommissioning (December 2018).	\$ 778.11 million
3.	A schedule of the annual amounts remaining to be collected.	None.
4.	Assumptions used in determining rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections.	Escalation rate: 3.0% Rate of earnings: 5.0%
5.	Any contracts upon which the licensee is relying pursuant to 10 CFR 50.75(e)(1)(v).	None.
6.	Modifications occurring to a licensee's current method of providing financial assurance since the last submitted report.	None.
7.	Any material changes to trust agreements.	None.

Attachment 1 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
Status of Decommissioning Funding
For Year Ending December 31, 2004 – 10 CFR 50.75(f)(1)

Plant Name: **Vermont Yankee Nuclear Power Station**

1.	Amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75 (b) and (c).	\$ 412.60 million [1]
	Decommissioning cost estimate escalated at 3.0% per year to the midpoint of decommissioning (December 2015).	\$ 571.13 million
2.	Amount accumulated to the end of the calendar year preceding the date of the report (December 31, 2004).	\$ 372.80 million
	Fund balance with 5.0% annual growth to the midpoint of decommissioning (December 2015).	\$ 637.61 million
3.	A schedule of the annual amounts remaining to be collected.	None.
4.	Assumptions used in determining rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections.	Escalation rate: 3.0% Rate of earnings: 5.0%
5.	Any contracts upon which the licensee is relying pursuant to 10 CFR 50.75(e)(1)(v).	None.
6.	Modifications occurring to a licensee's current method of providing financial assurance since the last submitted report.	None.
7.	Any material changes to trust agreements.	None.

Entergy Nuclear Operations, Inc.
Status of Decommissioning Funding
For Year Ending December 31, 2004 – 10 CFR 50.75(f)(1)

Plant Name: **Pilgrim Nuclear Power Station**

1.	Amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75 (b) and (c).	$ 426.25 million [1]
	Decommissioning cost estimate escalated at 3.0% per year to the midpoint of decommissioning (December 2015).	$ 590.03 million
2.	Amount accumulated to the end of the calendar year preceding the date of the report (December 31, 2004).	$ 528.74 million
	Fund balance with 5.0% annual growth to the midpoint of decommissioning (December 2015).	$ 904.32 million
3.	A schedule of the annual amounts remaining to be collected.	None.
4.	Assumptions used in determining rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections.	Escalation rate: 3.0% Rate of earnings: 5.0%
5.	Any contracts upon which the licensee is relying pursuant to 10 CFR 50.75(e)(1)(v).	None.
6.	Modifications occurring to a licensee's current method of providing financial assurance since the last submitted report.	None. [see item 7]
7.	Any material changes to trust agreements.	In March 2003, Mellon Bank, the trustee of the Pilgrim Provisional Decommissioning Trust was given direction to contribute all assets remaining in the Provisional Trust to the non-qualified fund of the Pilgrim Master Trust. Later that month, Mellon Bank transferred approximately $30 million of assets as directed. The Pilgrim Provisional Trust was then terminated.

Attachment 1 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
Status of Decommissioning Funding
For Year Ending December 31, 2004 – 10 CFR 50.75(f)(1)

Plant Name: James A. Fitzpatrick Nuclear Power Plant

1.	Amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75 (b) and (c).	$ 442.19 million [1]
	Decommissioning cost estimate escalated at 3.0% per year to the midpoint of decommissioning (December 2017).	$ 649.37 million
2.	Amount accumulated to the end of the calendar year preceding the date of the report (December 31, 2004).	$ 428.80 million
	Fund balance with 5.0% annual growth to the midpoint of decommissioning (December 2017).	$ 808.57 million
3.	A schedule of the annual amounts remaining to be collected.	None.
4.	Assumptions used in determining rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections.	Escalation rate: 3.0% Rate of earnings: 5.0%
5.	Any contracts upon which the licensee is relying pursuant to 10 CFR 50.75(e)(1)(v).	None.
6.	Modifications occurring to a licensee's current method of providing financial assurance since the last submitted report.	None.
7.	Any material changes to trust agreements.	None.

Entergy Nuclear Operations, Inc.
Status of Decommissioning Funding
For Year Ending December 31, 2004 – 10 CFR 50.75(f)(1)

Notes:

[1] The calculation of the NRC minimum value is provided in Attachment 2.

[2] The current fund balances for Indian Point 1 and 2 do not include an additional $25.91 million available in the provisional fund.

[3] In accordance with 10 CFR 50.75(c)(i)(1) PWR reactors below 1200 MWt are to use this minimum value. Indian Point 1 had a thermal power level of 615 MWt. (Refer to Attachment 3, pg. 15, of June 8, 2001 letter, M. R. Kansler to USNRC regarding "Response to June 5, 2001 Letter, Indian Point Nuclear Generating Unit Nos. 1 and 2, Transfer of Facility Operating License (TAC Nos. MB0743 and MB0744).")

Attachment 2 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
NRC Minimum Funding Calculation (10 CFR 50.75(c))
For Year Ending December 31, 2004

Plant Name: **Indian Point Nuclear Generating Unit No. 1**

Inputs:

Plant Characteristics

Plant Type (PWR or BWR)	PWR
Region	NE (Northeast)
Rated (in MWt) [1]	615
Year	2004
Waste Vendor Used (Yes/No)	Yes
Burial Site	South Carolina
Atlantic Compact Member (Yes/No)	No

Producer Price Index

wpu0543 (industrial electric power)	147.9 (December 2004)
wpu0573 (light fuel oils)	133.4

Labor Adjustment Factors

ecu13102i (Northeast)	174.2 (4th Quarter, 2004)

Burial Site Adjustments (South Carolina/non-Atlantic Compact Member)

PWR - direct disposal	18.732 (NUREG-1307, Rev. 10)
PWR - w/waste vendors	9.467

Adjustment Factors

Energy (E)	1.434
Labor (L)	2.076
Burial (B)	9.467

Minimum Amount (millions, $1986)	$85.56	(size adjusted for megawatts)
Escalation Factor (L, E, B)	3.618	(65% L, 13% E, 22% B)

NRC Minimum (millions, $2004)	**$309.59**	w/waste vendor

[1] In accordance with 10 CFR 50.75(c)(I)(1) PWR reactors below 1200 MWt are to use this minimum value. Indian Point 1 had a thermal power level of 615 MWt. (Refer to Attachment 3, pg. 15, of June 8, 2001 letter, M. R. Kansler to USNRC regarding "Response to June 5, 2001 Letter, Indian Point Nuclear Generating Unit Nos. 1 and 2, Transfer of Facility Operating License (TAC Nos. MB0743 and MB0744).")

Attachment 2 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
NRC Minimum Funding Calculation (10 CFR 50.75(c))
For Year Ending December 31, 2004

Plant Name: **Indian Point Nuclear Generating Unit No. 2**

Inputs:

Plant Characteristics

Plant Type (PWR or BWR)	PWR
Region	NE (Northeast)
Rated (in MWt)	3216
Year	2004
Waste Vendor Used (Yes/No)	Yes
Burial Site	South Carolina
Atlantic Compact Member (Yes/No)	No

Producer Price Index

wpu0543 (industrial electric power)	147.9 (December 2004)
wpu0573 (light fuel oils)	133.4

Labor Adjustment Factors

ecu13102i (Northeast)	174.2 (4th Quarter, 2004)

Burial Site Adjustments (South Carolina/non-Atlantic Compact Member)

PWR - direct disposal	18.732 (NUREG-1307, Rev. 10)
PWR - w/waste vendors	9.467

Adjustment Factors

Energy (E)	1.434
Labor (L)	2.076
Burial (B)	9.467

Minimum Amount (millions, $1986)	$103.30 (size adjusted for megawatts)	
Escalation Factor (L, E, B)	3.618 (65% L, 13% E, 22% B)	
NRC Minimum (millions, $2004)	**$373.79**	**w/waste vendor**

Attachment 2 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
NRC Minimum Funding Calculation (10 CFR 50.75(c))
For Year Ending December 31, 2004

Plant Name: **Indian Point Nuclear Generating Unit No. 3**

Inputs:

Plant Characteristics

Plant Type (PWR or BWR)	PWR
Region	NE (Northeast)
Rated (in MWt)	3067.4
Year	2004
Waste Vendor Used (Yes/No)	Yes
Burial Site	South Carolina
Atlantic Compact Member (Yes/No)	No

Producer Price Index

wpu0543 (industrial electric power)	147.9 (December 2004)
wpu0573 (light fuel oils)	133.4

Labor Adjustment Factors

ecu13102i (Northeast)	174.2 (4th Quarter, 2004)

Burial Site Adjustments (South Carolina/non-Atlantic Compact Member)

PWR - direct disposal	18.732 (NUREG-1307, Rev. 10)
PWR - w/waste vendors	9.467

Adjustment Factors

Energy (E)	1.434
Labor (L)	2.076
Burial (B)	9.467

Minimum Amount (millions, $1986)	$101.99 (size adjusted for megawatts)
Escalation Factor (L, E, B)	3.618 (65% L, 13% E, 22% B)

NRC Minimum (millions, $2004) **$369.06** w/waste vendor

Attachment 2 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
NRC Minimum Funding Calculation (10 CFR 50.75(c))
For Year Ending December 31, 2004

Plant Name: **Vermont Yankee Nuclear Power Station**

Inputs:

Plant Characteristics

Plant Type (PWR or BWR)	BWR
Region	NE (Northeast)
Rated (in MWt)	1593
Year	2004
Waste Vendor Used (Yes/No)	Yes
Burial Site	South Carolina
Atlantic Compact Member (Yes/No)	No

Producer Price Index

wpu0543 (industrial electric power)	147.9 (December 2004)
wpu0573 (light fuel oils)	133.4

Labor Adjustment Factors

ecu13102i (Northeast)	174.2 (4th Quarter, 2004)

Burial Site Adjustments (South Carolina/non-Atlantic Compact Member)

BWR - direct disposal	16.705 (NUREG-1307, Rev. 10)
BWR - w/waste vendors	8.860

Adjustment Factors

Energy (E)	1.448
Labor (L)	2.076
Burial (B)	8.860

Minimum Amount (millions, $1986)	$118.34 (size adjusted for megawatts)
Escalation Factor (L, E, B)	3.487 (65% L, 13% E, 22% B)

NRC Minimum (millions, $2004)	$\boxed{\$412.60}$	w/waste vendor

Attachment 2 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
NRC Minimum Funding Calculation (10 CFR 50.75(c))
For Year Ending December 31, 2004

Plant Name: **Pilgrim Nuclear Power Station**

Inputs:

Plant Characteristics

Plant Type (PWR or BWR)	BWR
Region	NE (Northeast)
Rated (in MWt)	2028
Year	2004
Waste Vendor Used (Yes/No)	Yes
Burial Site	South Carolina
Atlantic Compact Member (Yes/No)	No

Producer Price Index

wpu0543 (industrial electric power)	147.9 (December 2004)
wpu0573 (light fuel oils)	133.4

Labor Adjustment Factors

ecu13102i (Northeast)	174.2 (4th Quarter, 2004)

Burial Site Adjustments (South Carolina/non-Atlantic Compact Member)

BWR - direct disposal	16.705 (NUREG-1307, Rev. 10)
BWR - w/waste vendors	8.860

Adjustment Factors

Energy (E)	1.448
Labor (L)	2.076
Burial (B)	8.860

Minimum Amount (millions, $1986)	$122.25 (size adjusted for megawatts)
Escalation Factor (L, E, B)	3.487 (65% L, 13% E, 22% B)

NRC Minimum (millions, $2004) **$426.25** w/waste vendor

Attachment 2 to BVY-05-033, NL-05-039, JPN-05-005, ENO Ltr. 2.05.023

Entergy Nuclear Operations, Inc.
NRC Minimum Funding Calculation (10 CFR 50.75(c))
For Year Ending December 31, 2004

Plant Name: **James A. Fitzpatrick Nuclear Power Plant**

Inputs:

Plant Characteristics

Plant Type (PWR or BWR)	BWR
Region	NE (Northeast)
Rated (In MWt)	2536
Year	2004
Waste Vendor Used (Yes/No)	Yes
Burial Site	South Carolina
Atlantic Compact Member (Yes/No)	No

Producer Price Index

wpu0543 (industrial electric power)	147.9 (December 2004)
wpu0573 (light fuel oils)	133.4

Labor Adjustment Factors

ecu13102i (Northeast)	174.2 (4th Quarter, 2004)

Burial Site Adjustments (South Carolina/non-Atlantic Compact Member)

BWR - direct disposal	16.705 (NUREG-1307, Rev. 10)
BWR - w/waste vendors	8.860

Adjustment Factors

Energy (E)	1.448
Labor (L)	2.076
Burial (B)	8.860

Minimum Amount (millions, $1986)	$126.82 (size adjusted for megawatts)
Escalation Factor (L, E, B)	3.487 (65% L, 13% E, 22% B)

NRC Minimum (millions, $2004)	$\boxed{\$442.19}$	w/waste vendor

EXHIBIT Y

EXHIBIT Z

[Federal Register: August 1, 2007 (Volume 72, Number 147)]
[Notices]
[Page 42134-42135]
From the Federal Register Online via GPO Access [wais.access.gpo.gov]
[DOCID:fr01au07-109]

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50-247 and 50-286]

Entergy Nuclear Operations, Inc., Indian Point Nuclear Generating Unit Nos. 2 and 3; Notice of Acceptance for Docketing of the Application and Notice of Opportunity for Hearing Regarding Renewal of Facility Operating License Nos. DPR-26 and DPR-64 for an Additional 20-Year Period

The U.S. Nuclear Regulatory Commission (NRC or the Commission) is considering an application for the renewal of Operating License Nos. DPR-26 and DPR-64, which authorize Entergy Nuclear Operations, Inc., to operate Indian Point Nuclear Generating Unit Nos. 2 and 3, respectively, at 3216 megawatts thermal (MWt) for each unit. The renewed licenses would authorize the applicant to operate Indian Point Nuclear Generating Unit Nos. 2 and 3 for an additional 20 years beyond the period specified in the current licenses. The current operating licenses for Indian Point Nuclear Generating Unit Nos. 2 and 3 expire on September 9, 2013, and December 12, 2015, respectively.

Entergy Nuclear Operations, Inc. submitted the application dated April 23, 2007, as supplemented by letters dated May 3, 2007, and June 21, 2007, pursuant to 10 CFR Part 54, to renew Operating License Nos. DPR-26 and DPR-64 for Indian Point Nuclear Generating Unit Nos. 2 and 3, respectively. A Notice of Receipt and Availability of the license renewal application (LRA), ``Entergy Nuclear Operations, Inc.; Notice of Receipt and Availability of Application for Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3; Facility Operating Licenses Nos. DPR-26 and DPR-64 for an Additional 20-Year Period,'' was published in the Federal Register on May 11, 2007 (72 FR 26850).

The Commission's staff has determined that Entergy Nuclear Operations, Inc. has submitted sufficient information in accordance with 10 CFR Sections 54.19, 54.21, 54.22, 54.23, 51.45, and 51.53(c) to enable the staff to undertake a review of the application, and the application is therefore acceptable for docketing. The current Docket Nos. 50-247 and 50-286 for Operating License Nos. DPR-26 and DPR-64, respectively, will be retained. The determination to accept the license renewal application for docketing does not constitute a determination that a renewed license should be issued, and does not preclude the NRC staff from requesting additional information as the review proceeds.

Before issuance of each requested renewed license, the NRC will have made the findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. In accordance with 10 CFR 54.29, the NRC may issue a renewed license on the basis of its review if it finds that actions have been identified and have been or will be taken with respect to: (1) Managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified as requiring aging management review, and (2) time-limited aging analyses that have been identified as requiring review, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis (CLB), and that any changes made to the plant's CLB comply with the Act and the Commission's regulations.

Additionally, in accordance with 10 CFR 51.95(c), the NRC will prepare an environmental impact statement that is a supplement to the Commission's NUREG-1437, ``Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants,'' dated May 1996. In considering the license renewal application, the Commission must find that the applicable requirements of Subpart A of 10 CFR Part 51 have been satisfied. Pursuant to 10 CFR 51.26, and as part of the environmental scoping process, the staff intends to hold a public scoping meeting. Detailed information regarding the environmental scoping meeting will be the subject of a separate Federal Register notice.

Within 60 days after the date of publication of this Federal Register Notice, any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing or a petition for leave to intervene with respect to the renewal of the license. Requests for a hearing or petitions for leave to intervene must be filed in accordance with the Commission's ``Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852 and is accessible from the NRC's Agencywide Documents Access and Management

System (ADAMS) Public Electronic Reading Room on the Internet at http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to

ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC's PDR reference staff by telephone at 1-800-397-4209 or 301-415-4737, or by e-mail at pdr@nrc.gov. If a request for a hearing/petition for leave to intervene is filed within the 60-day period, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order. In the event that no request for a hearing or petition for leave to intervene is filed within the 60-day period, the NRC may, upon completion of its evaluations and upon making the findings required under 10 CFR Parts 51 and 54, renew the license without further notice.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding, taking into consideration the limited scope of matters that may be considered pursuant to 10 CFR Parts 51 and 54. The petition must specifically explain the

[[Page 42135]]

reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated in the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the requestor/petitioner shall provide a brief explanation of the bases of each contention and a concise statement of the alleged facts or the expert opinion that supports the contention on which the requestor/petitioner intends to rely in proving the contention at the hearing. The requestor/petitioner must also provide references to those specific sources and documents of which the requestor/petitioner is aware and on which the requestor/petitioner intends to rely to establish those facts or expert opinion. The requestor/petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact.\1\ Contentions shall be limited to matters within the scope of the action under consideration. The contention must be one that, if proven, would entitle the requestor/petitioner to relief. A requestor/petitioner who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

\1\ To the extent that the application contains attachments and supporting documents that are not publicly available because they are asserted to contain safeguards or proprietary information, petitioners desiring access to this information should contact the applicant or applicant's counsel to discuss the need for a protective order.

The Commission requests that each contention be given a separate numeric or alpha designation within one of the following groups: (1) Technical (primarily related to safety concerns); (2) environmental; or (3) miscellaneous.

As specified in 10 CFR 2.309, if two or more requestors/petitioners seek to co-sponsor a contention or propose substantially the same contention, the requestors/petitioners will be required to jointly designate a representative who shall have the authority to act for the requestors/petitioners with respect to that contention.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing. A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, HEARINGDOCKET@NRC.GOV; or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC., Attention: Rulemaking and Adjudications Staff at 301-415-1101 (verification number: 301-415-1966).\2\ A copy of the request for hearing or petition for leave to intervene must also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and it is requested that copies be transmitted either by means of facsimile transmission to 301-415-3725 or by e-mail to OGCMailCenter@nrc.gov. A copy of the request for hearing or petition

for leave to intervene should also be sent to the Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

\2\ If the request/petition is filed by e-mail or facsimile, an original and two copies of the document must be mailed within 2 (two) business days thereafter to the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; Attention: Rulemaking and Adjudications Staff.

Non-timely requests and/or petitions and contentions will not be entertained absent a determination by the Commission, the presiding officer, or the Atomic Safety and Licensing Board that the petition, request and/or contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(a)(1)(i)-(viii).

Detailed information about the license renewal process can be found under the Nuclear Reactors icon at http://www.nrc.gov/reactors/operating/licensing/renewal.html on the NRC's Web site. Copies of the

application to renew the operating licenses for Indian Point Nuclear Generating Unit Nos. 2 and 3, are available for public inspection at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852-2738, and at http://www.nrc.gov/reactors/operating/licensing/renewal/applications.html , the

NRC's Web site while the application is under review. The application may be accessed in ADAMS through the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov/reading-rm/adams.html under

ADAMS Accession Numbers ML071210507, ML071280700, and ML071800318. As stated above, persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS may contact the NRC Public Document Room (PDR) Reference staff by telephone at 1-800-397-4209 or 301-415-4737, or by e-mail to pdr@nrc.gov.

The NRC staff has verified that a copy of the license renewal application is also available to local residents near Indian Point Nuclear Generating Unit Nos. 2 and 3 at the White Plains Public Library, 100 Martine Avenue, White Plains, NY 10601; the Field Library, 4 Nelson Avenue, Peekskill, NY 10566; and the Hendrick Hudson Free Library, 185 Kings Ferry Road, Montrose, NY 10548.

Dated at Rockville, Maryland, this 25th day of July, 2007.

For The Nuclear Regulatory Commission.
Pao-Tsin Kuo,
Director, Division of License Renewal, Office of Nuclear Reactor Regulation.

[FR Doc. E7-14864 Filed 7-31-07; 8:45 am]

BILLING CODE 7590-01-P

UNITED STATES DISTRICT COURT
EASTERN DISTRICT OF NEW YORK
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IN RE ZYPREXA LITIGATION, No. 07 Civ. 0504 (JBW)

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MEMORANDUM OF LAW ON BEHALF OF DAVID EGILMAN, M.D., M.P.H., IN OPPOSITION TO ELI LILLY AND COMPANY'S JANUARY 31, 2007, MEMORANDUM OF POINTS AND AUTHORITIES

See page 13

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PRELIMINARY STATEMENT

This memorandum of law is submitted on behalf of Dr. David Egilman in response to petitioner Eli Lilly and Company's ("Eli Lilly" or "petitioner") memorandum of law in support of its request to modify and extend the Court's mandatory injunction entered January 3, 2007. As Eli Lilly concedes, and as has been noted by this Court, Dr. Egilman is not a party to the instant proceedings, nor is he a subject of petitioner's request for a mandatory injunction. (See Petitioner's Proposed Order for Mandatory Injunction .) Indeed, although Eli Lilly has threatened for more than a month to file charges of contempt against Dr. Egilman, no such charges have been forthcoming. Instead, Eli Lilly has mounted yet another attempt to erode Dr. Egilman's due process rights by asking this Court to prematurely decide issues of fact central to its anticipated contempt motion and irrelevant to its request for an injunction.[1] Specifically, in its papers currently before the Court, Eli Lilly requests that this Court make certain factual findings and legal conclusions which will inevitably prejudice Dr. Egilman's defense against the threatened contempt motion. For several reasons, petitioner's request should be denied.

There are four areas implicated by Eli Lilly's brief which, if decided in petitioner's favor, will have the effect of seriously prejudicing Dr. Egilman's ability to defend against a future contempt motion: (1) the request that the Court find that Dr. Egilman violated CMO-3 when he produced documents to Jim Gottstein in response to Mr. Gottstein's subpoena dated December 6, 2006; (2) Eli Lilly's contention that it acted in good faith when it designated every single one of the millions of documents produced in this matter as "confidential" and subject to the protective order at issue in this case, CMO-3; (3) petitioner request that the Court make certain findings of

[1] We have previously detailed Eli Lilly's prior attempts to use proceedings to which Dr. Egilman was not a party to bootstrap a finding of contempt against our client. (See January 16, 2007, Letter from Alexander A. Reinert to the Court, at 3-4; December 28, 2006, Letter from Alexander A. Reinert to the Court, at 3-4.)

fact regarding Dr. Egilman's actions and intentions which are not supported by admissible evidence and which have no bearing on petitioner's request for an injunction; and (4) Eli Lilly's request that the Court draw adverse factual inferences based on Dr. Egilman's invocation of his right to silence. Because neither the relevant law nor the admissible evidence supports Eli Lilly's premature requests, Dr. Egilman respectfully requests that the Court make no legal or factual findings which relate to Eli Lilly's anticipated motion for contempt against Dr. Egilman.

ARGUMENT

I. ELI LILLY HAS FAILED TO ESTABLISH THAT DR. EGILMAN ACTED IN CONTEMPT OF CMO-3

Eli Lilly asks this Court to adopt several factual findings to the effect that Dr. Egilman violated CMO-3 "in [c]oncert with [o]thers." (Eli Lilly And Co.'s Mem. Of Points And Authorities Concerning Its Request To Modify And Extend The Court's January 3, 2007 Temporary Mandatory Injunction ("Lilly Br.") at 4.) Indeed, Eli Lilly asks this court to find, without factual support, that Dr. Egilman "selectively leaked" documents to affect settlement discussions and "prejudice Lilly's right to a fair trial," (Proposed Revised Findings of Fact ("Proposed Findings") ¶¶ 18, 19), that Dr. Egilman agreed with Alex Berenson "on a scheme to bypass CMO-3 and get the protected documents to the *New York Times*," (id. ¶ 24), and that "Dr. Egilman had violated CMO-3 by sending Mr. Gottstein documents that he had received pursuant to the confidentiality provisions of CMO-3" (id. ¶ 42).[2] While petitioner does not explicitly ask this Court to find that Dr. Egilman acted in contempt of CMO-3, if the Court adopts Eli Lilly's

[2] One of the principal grounds for Eli Lilly's factual findings with regard to Dr. Egilman is his assertion of his Fifth Amendment right to silence. (See Proposed Findings ¶¶ 16-22, 27-28, 37 & n.1; Lilly Br. 4 n.3.) Unfortunately for petitioner, as detailed below, Dr. Egilman's assertion of his right to silence in this context does not

factual findings, it is hard to imagine what additional facts petitioner would present in support of its anticipated contempt motion. For several reasons, petitioner's request should be denied.

A. Eli Lilly Fails to Address the Standard of Proof for Establishing Contempt, and Accompanying Procedural Protections

As this Court is aware, any party seeking to establish contempt bears a "heavy" burden. See Air-Products and Chemicals, Inc. v. Inter-Chemicals, Ltd., No. 03 Civ. 6140, 2005 WL 196543, *3 (E.D. Pa. Jan. 27, 2005); cf. Perez v. Danbury Hosp., 347 F.3d 419, 423 (2d Cir. 2003) (district court's contempt power is "narrowly circumscribed," necessitating a more "exacting" appellate review than other discretionary decisions); see also Jan. 17, 2007 Tr., at 246 (noting that contempt is a "quagmire"). To establish civil contempt, petitioner bears the burden of presenting "clear and convincing" evidence that Dr. Egilman has violated a "clear and unambiguous" judicial order. Fonar Corp. v. Deccaid Servs., Inc., 983 F.2d 427, 429 (2d Cir. 1993); Hess v. New Jersey Transit Rail Operations, Inc., 846 F.2d 114, 115 (2d Cir.1988) (explaining that "[n]o one may be held in contempt for violating a court order unless the order is clear and specific and leaves no uncertainty in the minds of those to whom it is addressed"); Littlejohn v. BIC Corp., 851 F.2d 673, 683-84 (3d Cir. 1988); Lesco v. Masone, No. 05 Civ. 3207, 2006 WL 2166862, *5 (E.D.N.Y. July 27, 2006). To establish criminal contempt, which results in the imposition of punitive sanctions,[3] Eli Lilly must establish beyond a reasonable doubt that Dr. Egilman violated a similarly specific judicial order. Mackler Productions, Inc. v. Cohen, 225 F.3d 136, 142 (2d Cir. 2000); United States v. NYNEX Corp., 8 F.3d 52, 54 (D.C. Cir. 1993) (requiring willful violation of specific order).

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give Lilly free rein to imagine a set of facts which are "not inconsistent," (Lilly Br. 4 n.3) with evidence which has been submitted in this proceeding. See infra Part III.B.

[3] To be considered a "criminal" contempt proceeding, it is not necessary that a Court impose restrictions on the contemnor's liberty, such as jail or prison time. Indeed, fines as little as $1000 have been considered sufficient

Because of the serious nature of contempt charges, certain procedural protections must be satisfied prior to issuing a judgment of contempt. In civil contempt proceedings, the alleged contemnor is entitled to notice of the grounds for the contempt and an opportunity to be heard, but is not obliged to present any evidence should the movant fail in establishing any of the three elements of contempt. Perez, 347 F.3d at 423 (movant has burden of proof); Electrical Workers Pension Trust Fund v. Gary, 340 F.3d 373, 379 (6th Cir. 2003) (burden of proof and production rests with movant); see generally International Union, United Mine Workers of America v. Bagwell, 512 U.S. 821, 826-34 (1994) (discussing difference in protections provided for criminal versus civil contempt). In the criminal contempt context, even more protections apply, including "the right to a public trial, the assistance of counsel, the presumption of innocence, the privilege against self-incrimination, and the requirement of proof beyond a reasonable doubt." Mackler Productions, Inc. v. Cohen, 225 F.3d 136, 142 (2d Cir. 2000); see also Bloom v. Illinois, 391 U.S. 194 (1968) (criminal procedural protections apply to criminal contempt proceedings). In no event may the movant rely on unsworn assertions to establish the critical elements of civil or criminal contempt. Mackler Productions, 225 F.3d at 146 (it is error to rely on unsworn assertions to establish sanctions in contempt). And to support claims for damages, the movant must provide concrete evidence of harm. Mingoia v. Crescent Wall Sys., No. 03 Civ. 7143, 2005 WL 991773, *5 (S.D.N.Y. April 26, 2005).

For these reasons, courts have emphasized the importance early on of indicating whether the contempt proceedings are criminal or civil in nature, so that the appropriate procedural and substantive protections may be provided to the alleged contemnor. In re Jessen, 738 F. Supp. 960, 962 (W.D.N.C. 1990); cf. Ashcraft v. Conoco, Inc., 1998 WL 404491, *2 n.2 (E.D.N.C.,

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to trigger the due process protections associated with criminal prosecution. Hess v. New Jersey Transit Rail

Jan. 21, 1998), rev'd on other grounds, 218 F.3d 288 (4th Cir. 2000). This is because if the civil/criminal determination affects both the remedy that a court may impose and the procedural protections that must be followed throughout the proceeding. Hess v. New Jersey Transit Rail Operations, Inc., 846 F.2d 114, 115 (2d Cir. 1988) (sanction of as little as $1000 considered exercise of criminal contempt power because purpose was to punish for past violations).

B. Eli Lilly's Request that the Court Find that Dr. Egilman Violated CMO-3 Disregards Its Burden and Dr. Egilman's Due Process Rights

Eli Lilly ignores each of these basic principles in asking the Court to find that Dr. Egilman willfully violated CMO-3. For instance, although petitioner has consistently indicated that it seeks to obtain punitive sanctions for Dr. Egilman's alleged violation of CMO-3, (see, e.g., December 26, 2006, Letter from Nina M. Gussack to the Court, at 1), Eli Lilly seeks here to use Dr. Egilman's invocation of his right to silence to establish certain facts which will inevitably prejudice any future criminal contempt proceeding. (See Proposed Findings ¶¶ 16-22, 27-28, 37 & n.1; Lilly Br. 4 n.3.) In its haste to prematurely convince this Court that Dr. Egilman acted in contempt of CMO-3, Eli Lilly makes no effort to square its desire to pursue criminal sanctions against Dr. Egilman, which triggers the protections of the Fifth Amendment, Mackler Productions, 225 F.3d at 142, with its request that the Court hold Dr. Egilman's silence against him.[4]

Moreover, Eli Lilly has asked this Court to find that Dr. Egilman violated CMO-3 without even attempting to satisfy its burden of establishing that Dr. Egilman violated a "clear and specific" order. Hess, 846 F.2d at 115. After all, Eli Lilly concedes that Dr. Egilman complied with that part of CMO-3 which required him to notify petitioner that he had been

Operations, Inc., 846 F.2d 114, 115 (2d Cir. 1988).

served with a subpoena requesting disclosure of "confidential" documents. (Proposed Findings ¶ 38; Lilly Br. 5.) Therefore, Eli Lilly must establish that CMO-3 clearly and specifically communicated what constituted a "reasonable opportunity" for Eli Lilly to object to production of the documents. (Proposed Findings, Ex. 2, at ¶ 14.)

Eli Lilly cannot claim that it did not have notice of the anticipated disclosure. Nor can petitioner claim that Dr. Egilman's failure to notify Pepper Hamilton directly – a failing not of Dr. Egilman's making but of the protective order that had been drafted by Eli Lilly – prejudiced its rights, because it asserts that it "promptly" provided notice of the subpoena to its outside counsel, and that these counsel "took immediate steps to determine who had retained" Dr. Egilman.[5] (Lilly Br. 5.) Although Eli Lilly does not detail the steps it took, given the stated importance of the documents, one would expect that petitioner had a reasonable opportunity to make its objection known to Dr. Egilman before he disclosed the documents to Mr. Gottstein.[6] In any event, the factual record is simply insufficient at this time to make that determination, although we note in passing that in litigating the issues currently before the Court, Eli Lilly has adopted the position that one or two days' notice to address complex legal issues – such as the propriety of Orders to Show Cause or mandatory injunctions – is sufficient for other parties. (See, e.g., Eli Lilly's Proposed Order to Show Cause Issued to Dr. David Egilman (submitted on

[4] The significance of Dr. Egilman's Fifth Amendment rights in this proceeding is addressed more fully in Part III.B of this brief.

[5] Eli Lilly makes the bald assertion that it could not contact Dr. Egilman directly, because of his status as an expert, but it is far from clear that any ethical rule prohibited Mr. Armitage from contacting Dr. Egilman about a matter that had no connection to Dr. Egilman's consultation with The Lanier Law Firm. In any event, there was no arguable prohibition on contacting Mr. Gottstein, whose contact information was provided in the material sent to Mr. Armitage. (Jan. 17, 2007, Tr. 129.)

[6] Eli Lilly has refused to make its General Counsel available for a deposition to determine what steps were taken to make petitioner's objections known to Dr. Egilman. Eli Lilly appears to base its argument in favor of contempt on the ground that Richard Meadow testified that he contacted Dr. Egilman and informed him "not to do anything." (Proposed Findings, Ex. 3, ¶ 9.) Notwithstanding the fact that there is no evidence that Mr. Meadow told Dr. Egilman that Eli Lilly intended to move to prevent disclosure, or had moved to prevent disclosure, because

December 26, 2006, and calling for a response within 24 hours).)

Moreover, Dr. Egilman's actions with respect to the subpoena were consistent with the language of CMO-3, which only contained one defined time limit where an individual intends to disclose confidential documents. Specifically, CMO-3, drafted by Eli Lilly, provides only three days' notice to petitioner where disclosure of confidential information is to be made to a customer or competitor of Eli Lilly. (Proposed Findings, Ex. 2, at ¶ 6.). If, after three days has passed, no motion is filed objecting to the proposed disclosure, then CMO-3 is not violated. Given that Eli Lilly asserts that the principal purpose of the confidentiality designations in this case is to protect its competitive advantage, it defies reason for Eli Lilly to simultaneously argue that the notice provided in the instant case did not afford it sufficient opportunity to raise an objection to disclosure.

Even if the Court determines now that Eli Lilly did not have a reasonable opportunity to object to Dr. Egilman's disclosure, this does not end the inquiry, because petitioner must still show that CMO-3 specifically defines what is meant by "a reasonable opportunity to object." In contempt proceedings, whether criminal or civil, ambiguities must be resolved in "favor of the party charged with contempt." Air-Products and Chemicals, Inc. v. Inter-Chemicals, Ltd., No. 03 Civ. 6140, 2005 WL 196543, *3 (E.D. Pa. Jan. 27, 2005). For instance, where a protective order fails to spell out specific steps to be taken with protected documents, it is error to find even an attorney, judged to be better able to decipher ambiguous court orders, see United States v. Cutler, 58 F.3d 825, 835 (2d Cir. 1995), in contempt for revealing confidential information, Littlejohn v. BIC Corp., 851 F.2d 673, 686 (3d Cir. 1988). And where it is not apparent from the Order what kind of conduct is prohibited, then the Order will not be considered sufficiently

Mr. Meadow's telephone call occurred after a Eli Lilly had had a reasonable opportunity to object to disclosure, it clear and specific to support a contempt finding. NYNEX Corp., 8 F.3d at 55-57.

Notably, court orders which were more clear and specific than CMO-3 have been found insufficiently clear to justify contempt in other contexts. For instance, where a district court ordered a litigant to "stay out of the facilities up here on this floor unless you get prior permission. That's the jury room, also," this was not considered sufficiently clear and unambiguous to support contempt for a defendant who was found in the jury room after the Order was issued. United States v. O'Quinn, 913 F.2d 221, 222 (5th Cir. 1990) (per curiam). And the Second Circuit has held that the phrase "bonafide offer of settlement" is "vague and imprecise" and therefore will not support a conviction for contempt. Hess v. New Jersey Transit Rail Operations, Inc., 846 F.2d 114, 115 (2d Cir.1988). Just as in Hess, in which the Court could not decipher what kind of settlement offer would be a "bonafide" one, here it is impossible to state precisely what is meant by "reasonable opportunity to object" as used in CMO-3. Similarly, the Second Circuit has held that an order that prohibited a party from using certain software was insufficient to support contempt where the order "never defined precisely what constituted 'Maintenance Software.'" Fonar Corp. v. Deccaid Servs., Inc., 983 F.2d 427, 429 (2d Cir. 1993); see also John Paul Mitchell Sys. v. Quality King Distributors, Inc., No. 99 Civ. 9905, 2001 WL 910405, *7 (S.D.N.Y. Aug. 13, 2001) (order, which prevented Quality King from "movement, transfer, or other distribution" of a product, did not clearly prohibit company from purchasing and receiving products (and thereby "moving" the products to its own warehouse), nor did it prohibit company from aiding the sale of product by another company, because Order simply forbid the company "from itself distributing the products"). The Court noted that even if the defendant could be charged with actual knowledge of what the term

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has no bearing on the issue of Dr. Egilman's alleged contempt.

"Maintenance Software" meant, it would not matter, because the Order must be "clear and unambiguous" in a more objective sense. Fonar Corp., 983 F. 2d at 429. And this Circuit has seen numerous incidents in which even attorneys, expected to "comply with less specific orders than laymen," see Cutler, 58 F.3d at 835, have not been sanctioned for violating confidentiality, say, of settlement discussions, despite a clear prohibition of such disclosures. Calka v. Kucker Kraus & Bruh, 167 F.3d 144, 145 (2d Cir. 1999) (per curiam) (declining to impose sanction for attorney who violated confidentiality of CAMP proceedings); E-Z Bowz, L.L.C. v. Professional Prod. Res. Co., No. 00 Civ. 8670, 2003 WL 22416174, *3 (S.D.N.Y. Oct. 23, 2003) (declining to impose sanctions on attorney which violated court's confidentiality order regarding settlement discussions); Concerned Citizens of Bell Haven v. The Belle Haven Club, No. 99 Civ. 1467, 2002 WL 32124959, *5-6 (D. Conn. Oct. 25, 2002) (declining to impose sanction on party which publicly disclosed confidential settlement discussions).

The imprecision of CMO-3 is reinforced by those courts, in other legal contexts, that have noted the difficulty of interpreting the term "reasonable opportunity." Three Boys Music Corp. v. Bolton, 212 F.3d 477, 482 (9th Cir. 2000) (in copyright protection case, noting that "reasonable opportunity" is something more than "bare possibility," but distinguishing the two will sometimes "present a close question") (internal quotation marks omitted); Morgan v. Account Collection Technology, LLC, No. 05 Civ. 2131, 2006 WL 2597865, *4 (S.D.N.Y. Sept. 6, 2006) (noting that courts have not agreed on what constitutes a "reasonable opportunity" to file for class certification prior to a Rule 68 offer of judgment). To be sufficiently clear for purposes of contempt, there must have been "nothing ambiguous about [Dr. Egilman's] obligations" under CMO-3. JSC Foreign Economic Ass'n v. International Development & Trade Servs., Inc., No. 03 Civ. 5562, 2006 WL 1148110, *5 (S.D.N.Y. April 28, 2006). Eli Lilly has made no argument here that the CMO-3 unambiguously defined what constituted a "reasonably opportunity to object" to Dr. Egilman's disclosure of documents to Mr. Gottstein.[7] Instead, petitioner attempts to tar Dr. Egilman with testimony from Mr. Gottstein that he thought that Dr. Egilman thought that the documents should be publicly disseminated. (Jan. 16, 2007, Tr. at 26.) Putting aside the question of whether Mr. Gottstein's speculation amounts to clear and convincing evidence, or evidence beyond a reasonable doubt, of Dr. Egilman's motives, the CMO-3 does not prohibit individuals subject to the protective order from having a particular state of mind with respect to the disclosure of "confidential" documents; it prohibits disclosure of such documents without following the procedures laid out by CMO-3. If anything, however, Mr. Gottstein's testimony establishes that Dr. Egilman considered himself to be bound by the terms of the protective order, and sought to meet its terms as he interpreted them. (Id. at 27-30, 49-51; Jan. 17, 2007, Tr. at 75, 121-22, 133.) Whether Dr. Egilman simultaneously hoped that the documents would be publicly disseminated is irrelevant here, where Dr. Egilman did not violate a clear and specific court order upon disclosure of the documents to Mr. Gottstein. See United States v. NYNEX Corp., 8 F.3d 52, 55 (D.C. Cir. 1993) (in criminal contempt case, criticizing failure to distinguish between the need for violation to be willful and for the order to be clear and reasonably specific).

II. ELI LILLY HAS FAILED TO ESTABLISH ITS OWN COMPLIANCE WITH CMO-3

Aside from Eli Lilly's failure to establish that Dr. Egilman acted in contempt of CMO-3,

[7] Similarly, Eli Lilly's apparent contention that the CMO-3 "clearly and unambiguously," Fonar Corp., 983 F.2d at 429, called for Dr. Egilman to notify the Pepper Hamilton law firm rather than Eli Lilly itself is questionable. The CMO-3 refers nowhere to the Pepper Hamilton law firm and instead simply directs subpoena recipients to

petitioner also has failed to abide by this Court's direction that it provide an explanation for why the documents disclosed by Dr. Egilman should be considered "confidential." Paragraph 3 of CMO-3 defines confidential material as "any information that the producing party in good faith believes is properly protected under [Fed. R. Civ. P.] 26(c)(7)." (See Proposed Findings, Ex. 2.) Rule 26(c)(7), in turn, allows for the issuance of orders when necessary to protect "a trade secret or other confidential research, development, or commercial information." Fed. R. Civ. P. 26(c)(7). The Federal Rules recognize a "presumptive right of public access to discovery in all civil cases," absent a showing of good cause for a protective order pursuant to Rule 26(c). Loussier v. Universal Music Group, Inc., 214 F.R.D. 174, 176-177 (S.D.N.Y. 2003); see also In re "Agent Orange" Product Liability Litigation, 821 F.2d 139, 145-46 (2d Cir. 1989).[8] As the Supreme Court has recognized when drawing analogies between FOIA disclosures and the Federal Rules, "'there is no absolute privilege for trade secrets and similar confidential information.'" Federal Open Market Committee of Federal Reserve System v. Merrill, 443 U.S 340, 362 (1979) (quoting 8 C. Wright & A. Miller, Federal Practice and Procedure § 2043, p. 300 (1970).) Accordingly, the proponent of a protective order under Rule 26(c)(7) must show both that the confidentiality of trade secrets or commercial information is threatened, and that good cause exists to shield it from the public. Eli Lilly fails to satisfy either of these standards, despite the Court's direction at the January 17, 2007, proceedings to "be very specific" about "which of the documents that were exposed are documents, one, that constitute trade secrets or embarrassment of the other language under the rules and how their release has harmed you."

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contact the party responsible for the confidentiality designation, which Eli Lilly has conceded was done immediately by Dr. Egilman.

[8] In endorsing the protective order, in fact, the Court explicitly recognized that the public "has a right to know" about the subject documents, except to the extent that disclosure would undermine petitioner's ability to market Zyprexa in a competitive environment. (See Proposed Findings, Ex. 1, at 10-11.) The Court's concern was

(Jan. 17, 2007, Tr. at 242; see also id. at 243 ("[Y]ou are going to as quickly as possible tell them . . . which of the documents released you are going to specifically rely on, because I cannot, I believe, deal with the case on the ground that I know that in the millions of pages that we now have in our depository, there are some documents that should not have been released.").)

In New York, "[t]he subject of the trade secret must be secret" and "known only in the particular business in which it is used." DDS, Inc. v. Lucas Aerospace Power Transmission Corp., 182 F.R.D. 1, 4 (N.D.N.Y. 1998) (internal quotation marks and citation omitted). Federal courts in the Second Circuit look to six non-exclusive factors to determine whether information satisfies this test: "(1) the extent to which the information is known outside the business; (2) the extent to which it is known by employees and others involved in the business; (3) the measures taken to guard the information's secrecy; (4) the value of the information to the business or to its competitors; (5) the amount of time, money, and effort expended in development of the information; and (6) the ease or difficulty of duplicating or properly acquiring the information." Chembio Diagnostic Systems, Inc. v. Saliva Diagnostic Systems, Inc., 236 F.R.D. 129, 136 (E.D.N.Y. 2006). In satisfying this test, "there must be a specific demonstration of the facts as they relate to all of the factors, a specific articulation of the harm, and no reliance upon stereotypical and conclusory statements that the information is confidential." Sherwin-Williams Co. v. Spitzer, No. 04 Civ. 185, 2005 WL 2128938, *12 (N.D.N.Y. Aug. 24, 2005); Traveler's Ins. Co. v. Allied-Signal Inc. Master Pension Trust, 145 F.R.D. 17 (D. Conn. 1992) (denying motion for protective order for failure to "identify with particularity any serious harm that would result from public disclosure of the documents"). Moreover, as this Court has noted, "[a]n important factor in determining whether disclosure will cause competitive harm is whether the

not, as Eli Lilly seeks to imply, that psychiatric patients would somehow be harmed by disclosure. (See id.) If information that the party seeks to protect is current or stale." In re Agent Orange Product Liability Litigation, 104 F.R.D. 559, 575 (E.D.N.Y. 1985) (collecting cases indicating that information from one to 15 years old is entitled to less protection).

Eli Lilly has not sought to address these standards, or the Court's direction that it be "very specific" about which documents that were released are protected, because it simply cannot do so. The entirety of Eli Lilly's argument for the nondisclosure of the materials that were disclosed to Mr. Gottstein is based on a declaration from Gerald Hoffman, who reviewed an unspecified subset of documents more than one year ago. (Ex. A to Lilly Br. at 6.) Eli Lilly relies on Mr. Hoffman's stale declaration, arguing that there is "substantial overlap" between the documents reviewed by Mr. Hoffman and the documents disclosed by Dr. Egilman. (Lilly Br. 10 n.8.) This is not the specific identification or argument required by the Court, nor does it purport even to meet the Court's requirement that Eli Lilly be specific about how it has been harmed by the disclosure. Instead, Eli Lilly states without elaboration that the documents protect "confidential, and often draft or preliminary research and development information; strategic planning documents; employee training techniques; regulatory strategy; product development; competitor analyses; market research; potential marketing plans and strategies, or otherwise confidential material." (Lilly Br. 10.) And Mr. Hoffman, in turn, states that each of the documents "contains information related to: confidential research and development information; strategic plans; marketing plans, strategies; competitive analyses; market research; clinical trials and non-clinical trials; or interactions with key regulators or publishers" and "reveals something about Lilly's internal organization and structure, qualifies as intelligence data, and if disseminated would be useful to Lilly's competitors in the atypical antipsychotic marketplace."

anything, the continued suppression of the subject documents will harm psychiatric patients and their doctors.

(Ex. A to Lilly Br. ¶ 9.)

Notwithstanding Mr. Hoffman's broad brush strokes, however, even Eli Lilly appears to concede that disclosure of some of the documents he reviewed do not, as Mr. Hoffman previously swore under oath, constitute trade secrets or otherwise confidential business information. (Lilly Br. at 12 n. 10 (abandoning claim of confidentiality as to specific documents).) Thus, it is hard to take at face value Eli Lilly's attempt to rely upon his declaration now, when there is no evidence that he reviewed the documents disclosed by Dr. Egilman to Mr. Gottstein, or that his initial determination of confidentiality is reliable. Similarly, Eli Lilly can hardly argue that certain of the documents which have been identified – e.g., press releases and published newspaper articles – cannot be disclosed without disclosing confidential information. (See Proposed Findings, Ex. 11.)

Not only does Mr. Hoffman's declaration fail to meet this Court's direction from the Jauary 17, 2007, hearing, but it also fails to meet the burden of any movant seeking to establish that particular documents are eligible for protection under Rule 26(c)(7). For instance, that a party treats particular documents as confidential "does not mean they are automatically entitled to be subject to a protective order." Houbigant, Inc. v. Development Specialists, Inc., No. 01 Civ. 7388, 2003 WL 21688243, *2 (S.D.N.Y. Jul 21, 2003) (finding that documents relating to audit methodology are privileged, but requiring party to review documents to determine which are properly considered protected material). And reliance on conclusory assertions of confidentiality will doom an application for protective order under Rule 26(c)(7). See Salter v. I.C. System, Inc., No. 04 Civ. 1566, 2005 WL 3941662, *1 (D. Conn. May 3, 2005) (citing Cuno Inc. v. Pall Corp., 117 F.R.D. 506, 508 (E.D.N.Y.1987) (denying defendant's motion for protective order where "motion simply alleges that 'the documents have at all times been maintained as internal proprietary documents and contain valuable confidential technical information'")).

Moreover, the standard for businesses seeking a protective order is high. While natural persons may seek a protective order to protect against embarrassment, "protective orders are not available . . . to protect businesses from annoyance or embarrassment," unless the harm can be shown to have a specific monetizable value. Wilcock v. Equidev Capital L.L.C., No. 99 Civ. 10781, 2001 WL 913957, *1 (S.D.N.Y. Aug. 14, 2001) (names of businesses' clients are not confidential, based solely on speculation that "contacts with clients would be embarrassing or would result in the loss of those clients"). Conclusory allegations of harm, "unsubstantiated by specific examples or articulated reasoning, do not satisfy the Rule 26(c) test." Application of Akron Beacon Journal, No. 94 Civ. 1402, 1995 WL 234710, *10 (S.D.N.Y. April 20, 1995) (internal quotation marks omitted). Businesses must make a "specified showing of significant harm" to the business' competitive position; otherwise the good cause requirement would be "effectively undermine[d] . . ., as businesses would be able to claim good cause for any internal documents that portrayed the business in an unflattering light." Id. at *11. Even if certain documents "might injure [Eli Lilly's] commercial standing, that does not mean the are entitled to protection under a Rule 26(c) protective order. Littlejohn v. BIC Corp., 851 F.2d 673, 685 (3d Cir. 1988). As the Third Circuit has noted, "because release of information not intended by the writer to be for public consumption will almost always have some tendency to embarrass, an applicant for a protective order whose chief concern is embarrassment must demonstrate that the embarrassment will be particularly serious." Cipollone v. Liggett Group, Inc., 785 F.2d 1108, 1121 (3d Cir. 1986). Therefore, to the extent that Mr. Hoffman seeks to protect against "damaging Lilly's reputation and bolstering competitors' market shares," (Ex. A to Lilly Br. ¶

22), this is not a legitimate grounds for finding a document confidential.[9]

III. MANY OF THE FACTUAL FINDINGS PROPOSED BY ELI LILLY SHOULD BE REJECTED

A. Eli Lilly Asks This Court to Make Factual Findings Regarding Dr. Egilman Which Are Unsupported By The Record

Many of the factual findings that Eli Lilly asks this Court to make are based exclusively on Dr. Egilman's invocation of his Fifth Amendmen rights. (Proposed Findings ¶¶ 16-21.) We explain below, infra Part III.B, why petitioner's reliance upon Dr. Egilman's silence is misplaced. Even as to those factual findings which petitioner claims are supported by other evidence, however, they reflect faulty reasoning. Eli Lilly asks this Court to find, for instance, that Dr. Egilman executed an endorsement of CMO-3 prior to receiving Zyprexa documents from The Lanier Law Firm, (Proposed Findings ¶ 9) although no documentary evidence has been introduced to support this assertion. Similarly, there is no evidence that Dr. Egilman was informed of the multiple protective orders entered in this case prior to receiving the Zyprexa documents. (Id. ¶ 8.) Instead, the evidence presented by petitioner establishes only that Dr. Egilman executed a modified endorsement of CMO-3 after he received the Zyprexa documents. (Compare Proposed Findings, Exs. B & C to Ex. 3 with Proposed Findings ¶ 10.)

In addition, many of petitioner's proposed factual findings can only be supported by inadmissible hearsay. Petitioner's proposal that Dr. Egilman "needed to find a way" to transfer

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[9] Nor do the published newspaper articles in which the contents of Eli Lilly's documents have been revealed offer a compelling argument for why Eli Lilly designated as confidential the released documents. These articles have focused principally on two aspects of the documents released by Dr. Egilman to Mr. Gottstein: (1) Eli Lilly's awareness, and suppression, of information linking weight gain and diabetes to Zyprexa; and (2) Eli Lilly's illegal off-label marketing of Zyprexa for non-FDA-approved uses. Documents which reveal these facts ought not be protected by a Rule 26(c) Order. See Jack B. Weinstein, *Secrecy in Civil Trials: Some Tentative Views*, 9 J.L. & Pol'y 53, 61 (2000) (secrecy order which protects "those who engage in misconduct, conceal the cause of injury

documents to the New York Times, for instance, appears to be loosely based upon vague hearsay statements attributed by Mr. Gottstein to Alex Berenson. (Proposed Findings ¶ 22, citing Jan. 17, 2007, Tr. 96.) Similarly, Eli Lilly's proposal that Dr. Egilman and Mr. Berenson "agreed on a scheme to bypass CMO-3" either because Dr. Egilman understood that the documents were properly protected or because of concerns about timing of the release is remarkable, given that it is not supported by the portion of the transcript to which petitioner refers, which contains only inadmissible hearsay testimony. (Id. ¶ 24, citing Jan. 17, 2007, Tr. 96-98.) And petitioner's assertion that "Mr. Berenson told Dr. Egilman to contact Mr. Gottstein . . . and use him as the conduit for getting the protected documents to The New York Times" is similarly based on inadmissible hearsay. (Id. ¶ 25.)

Moreover, many of petitioner's proposed findings of fact are not supported by anything but rabid conjecture. For instance, there is no support in the material referenced in Proposed Revised Findings of Fact ¶ 30 for the proposition that Dr. Egilman played a role in determining what case Mr. Gottstein initiated prior to issuing the subpoena. Similarly, nothing in Mr. Gottstein's December 17, 2006, letter supports the inference that "he and Dr. Egilman worked in concert to issue a secret 'amended' subpoena," (Proposed Findings ¶ 47), nor did Eli Lilly elicit any testimony from Mr. Gottstein that he colluded with Dr. Egilman to issue the second subpoena. And accepting arguendo the dubious admissibility of Mr. Gottstein's testimony as to Dr. Egilman's understanding and intentions, Mr. Gottstein never testified that Dr. Egilman "understood and intended" that Mr. Gottstein would distribute the documents "as quickly as possible." (Proposed Findings ¶ 60.) Similarly, the excerpts of testimony relied upon for Proposed Revised Factual Finding ¶ 63 refer only to Mr. Gottstein's understanding, not Dr.

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from the victims, or render potential victims vulnerable . . . defeats a function of the justice system - to reveal

Egilman's.

B. Eli Lilly Improperly Seeks to Use Dr. Egilman's Invocation of His Fifth Amendment Right to Silence

Perhaps more troubling than petitioner's treatment of the record in this case is Eli Lilly's attempt to rely in this proceeding on Dr. Egilman's invocation of his right to silence. The Fifth Amendment permits adverse inferences against (1) parties when (2) "they refuse to testify in response to probative evidence offered against them." Mitchell v. United States, 526 U.S. 314, 327 (1999). Eli Lilly, on the other hand, seeks to use Dr. Egilman's invocation of his right to silence against other parties, and not based on Dr. Egilman's refusal to answer specific questions supported by probative evidence, but as a freewheeling opportunity for Eli Lilly to paint whatever picture it deems to be "not inconsistent" with the facts that have been presented. (See Proposed Findings ¶¶ 16-22, 27-28, 37 & n.1; Lilly Br. 4 n.3.). This poses several problems which are not acknowledged by petitioner.

1. Dr. Egilman's Silence Cannot Be Held Against the Respondents in This Proceeding

First, because Eli Lilly seeks to use Dr. Egilman's invocation of his right to silence against others, it must demonstrate a sufficiently close relationship between Dr. Egilman and the parties to be enjoined. The Second Circuit has held that, in determining whether a non-party's invocation of a Fifth Amendment privilege may be used to draw an adverse inference against a party to a proceeding, four non-exclusive factors should be considered: (1) the nature of the relationships between the party and non-party; (2) the "degree of control which the party has vested in the non-party witness in regard to the key facts" at issue in the litigation; (3)

important legal factual issues to the public").

convergence of interests between party and non-party; and (4) whether non-party was a "key figure in the litigation and played a controlling role in respect to any of its underlying aspects." LiButti v. United States, 107 F.3d 110, 123-24 (2d Cir. 1997). The fundamental question in undertaking this inquiry is "whether the adverse inference is trustworthy under all of the circumstances and will advance the search for the truth." Id. at 124.

Using this analysis, courts have held that an ex-employees' claim of privilege could be held against an employer as "vicarious admissions," id. at 121 (internal quotation marks omitted), and that a father's invocation of his Fifth Amendment privilege is admissible against his daughter where the relationship was strong and where they had "precisely the same interest," id. at 124. But where there is no formal or informal relationship with a non-party who has asserted his Fifth Amendment privilege, courts have permitted no adverse inference to be drawn even if the non-party has played a central role in the case. Willingham v. County of Albany, 2006 WL 1979048, *5 (N.D.N.Y. July 12, 2006). And even if the relationship has been shown to be very strong, if one party has chosen to subject herself to deposition, but the non-party has not, courts have held that that in and of itself demonstrates a sufficient disparity of interest to refuse to hold the non-party's assertion of a Fifth Amendment privilege against the party which agreed to testify. Kontos v. Kontos, 968 F. Supp. 400, 408 (S.D. Ind. 1997) (one sister's invocation of Fifth Amendment privilege inadmissible against other sister).

Here, Eli Lilly has made no attempt to establish, with respect to any of the four LiButti factors, that Dr. Egilman is sufficiently entwined with the parties to this proceeding such that his silence should be used to draw adverse inferences here. The instant proceeding seeks the continuation of an injunction only against the natural persons Terri Gottstein, Dr. Peter Breggin, Dr. David Cohen, Bruce Whittington, Laura Ziegler, Judi Chamberlin, Vera Sharav, Robert

Whittaker, Will Hall, Eric Whalen, and David Oaks, and against the websites www.joysoup.net, www.mindfreedom.org, www.ahrp.org, www.ahrp.blogspot.com, and zyprexa.pbwiki.com. Whatever the role that Eli Lilly argues that Dr. Egilman has played in this litigation, petitioner has not shown that he communicated with any of the parties to this proceeding, nor has Dr. Egilman been shown to be have any formal or informal relationship with them. Therefore, it is simply inappropriate for Eli Lilly to attempt to use his invocation of his right to silence against the parties to this proceeding. Lorusso v. Borer, No. 03 Civ. 504, 2006 WL 473729, *11 n.9 (D. Conn. Feb 28, 2006) (finding that it was inappropriate to use invocation of Fifth Amendment privilege against third party where there was "neither an agency relationship between the party and non-party concerned, nor any other close or intimate association between them, nor even a congruence of interests").

2. The Court Should Exercise Its Discretion to Reject Eli Lilly's Request for an Adverse Inference

Even were the Court to conclude that Dr. Egilman's relationship with the third parties against whom Eli Lilly seeks an injunction was sufficiently close, the Court still should exercise its discretion not to find an adverse inference from his invocation of his Fifth Amendment rights. Here, because Eli Lilly has indicated that it will seek criminal sanctions against Dr. Egilman, the Court should exercise its discretion not to rely in any way on Dr. Egilman's invocation of his Fifth Amendment rights. In general, where a party is simultaneously forced to defend against criminal and civil liability, courts take pains to protect Fifth Amendment rights by staying the civil proceeding until the criminal proceeding is complete so that the party will not be forced to choose between asserting his Fifth Amendment rights and fully defending himself in the civil action. For instance, in Sampson v. City of Schenectady, 160 F.Supp.2d 336 (N.D.N.Y. 2001), the Court declined to draw an adverse inference from a defendant's invocation of the Fifth

Amendment in response to a question at a deposition. The Court first noted that the drawing of an adverse inference "is a harsh remedy that is normally employed to counter a defendant's desire to obstruct discovery or abuse the privilege against self-incrimination." Id. at 351.[10] The Court found that, because the defendant had stated that he would go forward with a deposition once his criminal trial concluded, it would be unduly harsh to use his invocation of his Fifth Amendment rights against him in the civil proceeding. Id.

Of course, here, because of the unique framework of contempt motions, which permits litigants to seek criminal and civil remedies in the same proceeding, Eli Lilly has put Dr. Egilman in an untenable situation. Clearly petitioner seeks to establish certain facts in this proceeding which will later be used against Dr. Egilman in a hybrid criminal/civil contempt proceeding. In such a circumstance, it would be manifestly unfair for the Court to rely on Dr. Egilman's invocation of his constitutional right to silence to find facts which later will be used by Eli Lilly to support both criminal and civil penalties. In any event, just like the defendant in Sampson, here Dr. Egilman would be in a different position if criminal contempt proceedings were initiated first, or not at all, so that he could freely defend himself in the civil contempt proceeding.

This case is emblematic of the problem where one party has the opportunity to "manipulat[e] simultaneous civil and criminal proceedings," thus exacerbating "the risk to individuals' constitutional rights." Sterling Nat. Bank v. A-1 Hotels Intern., Inc., 175 F. Supp. 2d 573, 578 (S.D.N.Y. 2001). The Second Circuit identified this "dilemma" for litigants who are accused in parallel civil and criminal proceedings, such as forfeiture proceedings, noting that

[10] The Court also relied on the fact that the defendant had answered the civil complaint and denied certain allegations. Id. at 351. Of course, here, where Eli Lilly has yet to file a motion seeking contempt, Dr. Egilman has not had an opportunity to file any responsive pleading.

district courts must make "special efforts" in such cases to preserve both the Fifth Amendment privilege and the interest of all parties in the "opportunity to litigate a civil case fully." United States v. Certain Real Property and Premises Known as: 4003-4005 5th Ave., 55 F.3d 78, 83-84 (2d Cir. 1995). The options available to district courts include "entry of a protective order prohibiting the use of the civil litigant's responses in any criminal proceeding in that district, a stay of discovery or the civil action until parallel criminal proceedings have run their course, and attempts to arrange for immunity to ensure that a civil litigant will not be prosecuted for his or her statements." Id. at 84 n.6 (citations omitted). Courts in this Circuit have even granted a parties' request to stay the deadline for filing a responsive pleading in a civil case, where a parallel criminal proceeding remained open. Philip Morris Inc. v. Heinrich, No. 95 Civ. 0328, 1998 WL 167333, *1 (S.D.N.Y. April 8, 1998) ("If the Court orders Siegel to answer the Complaint and he asserts his Fifth Amendment privilege in his answer, he would be forced to suffer the consequence that a trier of fact may draw an adverse inference from his assertion of the privilege.")

Where, as with the anticipated contempt motion against Dr. Egilman, there are "overlapping issues in the criminal and civil cases, . . . the risk of impairing the party's Fifth Amendment rights presents a stronger case for staying discovery." American Express Business Finance Corp. v. RW Professional Leasing Servs. Corp., 225 F. Supp. 2d 263, 265 (E.D.N.Y. 2002); Savalle v. Kobyluck, Inc., No. 00 Civ. 675, 2001 WL 1571381, *2 (D. Conn. Aug. 15, 2001) ("Although requiring a defendant to choose between waiving his Fifth Amendment rights and suffering the adverse inference which results in the civil case from invoking his privilege does not violate due process, forcing the defendant to make this choice greatly increases the potential prejudice facing him in the absence of a stay."). This would prevent the party from being in "the uncomfortable position between choosing to waive their Fifth Amendment privilege, risking self-incrimination, or to invoke it, not only preventing them from adequately defending their position but subjecting them to adverse inferences." American Express, 225 F. Supp. 2d at 265. However, here, where Eli Lilly has not even begun proceedings against Dr. Egilman, but is seeking to lay an established factual foundation for its anticipated contempt motion, the Court should exercise its discretion to refuse to draw any adverse inference from Dr. Egilman's invocation of his Fifth Amendment rights. When Eli Lilly finally brings its contempt motion, the Court should then address whether to consider petitioner's request for criminal sanctions first, thus eliminating the pressure on Dr. Egilman to choose between asserting his constitutional rights and fully defending himself in the civil proceeding. For instance, even where a court was convinced that a party's "dilatory tactics in resolving his criminal case for more than two years" cautioned against continuing a stay of a parallel civil proceeding, the court nonetheless protected the party's Fifth Amendment right by giving the party "the opportunity to revoke the privilege prior to trial [in the civil proceeding] in order to avoid the adverse inference." Savalle, 2001 WL 1571381, at *3. Dr. Egilman should have the opportunity to argue that similar measures be taken when Eli Lilly finally institutes its contempt proceedings.

3. Eli Lilly Seeks to Rely on an Adverse Inference to Establish Facts That Are Not Supported By Any Other Independent Evidence

Even were the Court to be satisfied that due process does not forbid an adverse inference to be drawn in this proceeding from Dr. Egilman's invocation of his Fifth Amendment rights, Eli Lilly goes beyond all reasonable bounds by suggesting that the inference may support completely new facts which are "not inconsistent" with other facts adduced through this proceeding. It is well-established that the adverse inference requested by Eli Lilly may only be relied upon to "confirm matters supported by other independent evidence" and that liability may not be imposed "based solely upon the adverse inference." United States v. Incorporated Village of Island Park, 888 F. Supp. 419, 431-32 (E.D.N.Y. 1995). Here, no specific questions have been put to Dr. Egilman to which he has raised a Fifth Amendment objection; instead he has simply exercised his right to silence in the face of promises by Eli Lilly to seek criminal sanctions for his alleged violation of CMO-3.

Without relying on any other independent evidence, however, petitioner asks this Court to make certain factual findings based solely on Dr. Egilman's invocation of his Fifth Amendment rights. One must remember that the overriding purpose of reliance on the adverse inference in the Fifth Amendment context is the advancement of truth. LiButti, 107 F.3d at 123-24. Thus, an adverse inference is only appropriate where the Fifth Amendment is asserted in response to the presentation of "probative evidence." Mitchell, 526 U.S. at 327. Where a party seeks to draw adverse inferences with a sweeping scope, and with no support from corroborating admissible evidence, it cannot be said that relying on the adverse inference alone is likely to advance the search for truth. Emerson v. Wembley USA Inc., 433 F. Supp. 2d 1200, 1213 (D. Colo. 2006) (refusing adverse inference where plaintiff sought an inference based on all questions the witness refused to answer). This is especially the case where the inference has no foundation in any other evidence, and Eli Lilly is simply using a "fruitless deposition as a crutch to prop up [its] claims." Id. at 1214.

As this Court indicated when counsel first stated that Dr. Egilman intended to assert his Fifth Amendment right to silence at the deposition proposed by Eli Lilly, to the extent an adverse inference can be drawn, it should be limited to issues of credibility. (Jan. 17, 2007, Tr. 247); see also Orena v. United States, 956 F. Supp. 1071, 1094 (E.D.N.Y. 1997) ("Lack of credibility as a witness is not, in itself, affirmative proof of what the witness denies."). Dr. Egilman's credibility, however, is not relevant to the instant proceeding – there is no dispute that he received the subpoena from Mr. Gottstein, that he communicated it to Eli Lilly, and that he disclosed documents to Mr. Gottstein in response to the subpoena. The question in this proceeding is whether the injunction prohibiting further dissemination of these documents should be continued against certain individuals, who have no connection to Dr. Egilman. In such a circumstance, there is no basis for relying on Dr. Egilman's invocation of his right to silence to support petitioner's request that the injunction continue.

CONCLUSION

For the foregoing reasons, Dr. Egilman respectfully requests that the Court make no legal or factual findings which relate to Eli Lilly's anticipated motion for contempt against Dr. Egilman.

Dated: New York, New York
February 7, 2007

Respectfully submitted,

KOOB & MAGOOLAGHAN

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EXHIBIT AA

Introduced: Feb 12, 2007
Status: Introduced

Sponsor:
Rep. John Hall [D-NY]

H.R. 994: To require the Nuclear Regulatory Commission to conduct an Independent Safety Assessment of the...

HR 994 IH

110th CONGRESS

1st Session

H. R. 994

To require the Nuclear Regulatory Commission to conduct an Independent Safety Assessment of the Indian Point Energy Center.

IN THE HOUSE OF REPRESENTATIVES

February 12, 2007

Mr. HALL of New York (for himself, Mr. HINCHEY, Mr. ENGEL, Mrs. LOWEY, and Mr. SHAYS) introduced the following bill; which was referred to the Committee on Energy and Commerce

---

A BILL

To require the Nuclear Regulatory Commission to conduct an Independent Safety Assessment of the Indian Point Energy Center.

Be it enacted by the Senate and House of Representatives of the United States of America in Congress assembled,

SECTION 1. INDEPENDENT SAFETY ASSESSMENT.

Not later than 6 months after the date of enactment of this Act, the Nuclear Regulatory Commission shall transmit to the Congress a report containing the results of--

(1) a focused, in-depth Independent Safety Assessment of the design, construction, maintenance, and operational safety performance of the systems at the Indian Point Energy Center, Units 2 and 3, located in Westchester County, New York, including the systems described in section 2; and

(2) a comprehensive evaluation of the radiological emergency plan for Indian Point Energy Center, Units 2 and 3, conducted by the Nuclear Regulatory Commission and the Department of Homeland Security, which shall include--

(A) a detailed explanation of the factual basis upon which the Nuclear Regulatory Commission and the Federal Emergency Management Agency relied in--

(i) approving the radiological emergency plan; and

(ii) making subsequent annual findings of reasonable assurance that the plan will adequately protect the public in the event of an emergency, beginning on July 25, 2003 and continuing to the present;

(B) a detailed response to each of the criticisms of the radiological emergency plan contained in the Review of Emergency Preparedness of Areas Adjacent to Indian Point and Millstone, published by James Lee Witt Associates on January 10, 2003; and

(C) a detailed explanation of what criteria the Nuclear Regulatory Commission and Department of Homeland Security use in determining whether or not reasonable assurance can be provided that the radiological emergency plan is adequate to protect public health and safety, including what threshold figures of injuries and fatalities these agencies consider acceptable or tolerable in the event of a nuclear accident.

SEC. 2. SYSTEMS.

The systems referred to in section 1(1) are the following:

(1) The reactor protection system.

(2) The control room ventilation system and the containment ventilation system.

(3) The 4.16 kv electrical system.

(4) The condensate system.

(5) The spent fuel storage systems.

SEC. 3. INDEPENDENT SAFETY ASSESSMENT TEAM.

The Independent Safety Assessment conducted at Indian Point Nuclear Power Plant shall be conducted by an Independent Safety Assessment Team with 25 members, comprised of--

(1) 16 members from the Nuclear Regulatory Commission who are unaffiliated with the Nuclear Regulatory Commission Region 1 office or the Nuclear Regulatory Commission Office of Nuclear Reactor Regulation;

(2) 6 independent contractors with no history of having worked for or at the Indian Point Energy Center or any other nuclear power plant owned or operated by Entergy Corporation;

(3) the President of New York State Energy and Research Development Authority or his designee;

(4) the Director of the Bureau of Hazardous Waste and Radiation Management, in the Division of Solid and Hazardous Materials of the New York State Department of Environmental Conservation, or his designee; and

(5) a New York State-appointed independent contractor with experience in system engineering and no history of affiliation with any nuclear power plant owned by Entergy Corporation.

SEC. 4. INDEPENDENT SAFETY ASSESSMENT MONITORING.

The Independent Safety Assessment conducted at Indian Point Nuclear Energy Center shall be monitored by--

(1) an Independent Safety Assessment Observation Group comprised of 4 officials appointed by the State of New York; and

(2) an Independent Safety Assessment Citizens' Review Team comprised of 5 individuals appointed by the State of New York, with one resident from each Emergency Planning Zone county (Westchester, Rockland, Putnam, and Orange) appointed in consultation with the respective County Executive.

The Independent Safety Assessment Observation Group and Independent Safety Assessment Citizens' Review Team shall frequently provide publicly available updates on the progress and conduct of the Independent Security Assessment to the Governor of New York.

SEC. 5. INDEPENDENT SAFETY ASSESSMENT MODEL.

The Independent Safety Assessment conducted at Indian Point Energy Center shall be equal in scope, depth, and breadth to the Independent Safety Assessment of the Maine Yankee Nuclear Power Plant, located near Bath, Maine, conducted by the Nuclear Regulatory Commission in 1996.

SEC. 6. INCORPORATION INTO RELICENSING PROCESS.

The final decision by the Nuclear Regulatory Commission as to whether to renew the operating licenses for Unit 2 or Unit 3 at the Indian Point Energy Center shall not be made until--

(1) the Nuclear Regulatory Commission has fully entered the complete report and findings of the Independent Safety Assessment into the administrative record of the license renewal proceeding for Unit 2 and Unit 3 at the Indian Point Energy Center; and

(2) the applicant has fully accepted and implemented all findings and recommendations of the Independent Safety Assessment, including--

(A) undertaking all recommended repairs;

(B) replacement of safety-related equipment;

(C) changes to monitoring plans; and

(D) revision of the radiological emergency preparedness plans as called for in the report.

The applicant shall not be allowed to operate the reactors past the expiration date of its current operating licenses for Unit 2 and Unit 3 through administrative license renewals or any other means prior to meeting the requirements in paragraph (1) and paragraph (2) of this section.

SEC. 7. AUTHORIZATION OF APPROPRIATIONS.

There are authorized to be appropriated to the Nuclear Regulatory Commission to carry out this Act $10,000,000 for fiscal year 2008, to remain available until expended.

EXHIBIT BB

August 14, 2007 - 72 FR 45466-45467 - NUCLEAR REGULATORY COMMISSION - **Notice of Availability of the Final License Renewal Interim Staff Guidance LR-ISG-2006-03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses** - NRC is issuing its Final License Renewal Interim Staff Guidance LR-ISG-2006-03 for preparing severe accident mitigation alternatives (SAMA) analyses. This LR-ISG recommends that applicants for license renewal use the Guidance Document Nuclear Energy Institute 05-01, Revision A, (ADAMS Accession No. ML060530203) when preparing their SAMA analyses. The NRC staff issues LR-ISGs to facilitate timely implementation of the license renewal rule and to review activities associated with a license renewal application. The NRC staff will also incorporate the approved LR-ISG into the next revision of Supplement 1 to Regulatory Guide 4.2, ``Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses.'

EXHIBIT CC

PE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

May 2, 2003

NRC REGULATORY ISSUE SUMMARY 2003-09 ENVIRONMENTAL QUALIFICATION OF LOW-VOLTAGE INSTRUMENTATION AND CONTROL CABLES

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform addressees of the results of the technical assessment of GSI-168, "Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables." The scope of GSI-168 is limited to safety-related, low-voltage I&C cables. This RIS requires no action or written response on the part of an addressee.

BACKGROUND

In support of the resolution of GSI-168, the NRC sponsored cable test research at Wyle Laboratories and the Brookhaven National Laboratory. The resulting NRC technical assessment was essentially based on reviews and analyses of the research results of six loss-of-coolant-accident (LOCA) cable tests, condition-monitoring tests on I&C cables, and information provided by the nuclear industry. Technical assessments were coordinated with the nuclear industry and the Institute of Electrical and Electronics Engineers.

Following the completion of the NRC research effort, the staff concluded that typical I&C cable qualification test programs include numerous conservative practices that collectively provide a high level of confidence that the installed I&C cables will perform their intended functions during and following design basis events as required by 10 CFR 50.49, "Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants." These conservative practices continue to support the current use of a single prototype during qualification testing and, therefore, a successful test provides a high level of confidence that these cables will be able to perform their safety functions during and following a design basis event. However, cable LOCA test failures that occurred during the NRC-sponsored research program indicate that in certain cases the original margin and conservatism inherent in the qualification process have been reduced. Licensees have stated in a few cases that a reduction in margin can be addressed by monitoring operating service environments (temperature, radiation, and humidity)

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to ensure that operating conditions do not exceed the parameters that were assumed during qualification testing. In this regard, walkdowns to look for any visible signs of anomalies attributable to aging, with particular emphasis on localized adverse environments, coupled with the knowledge of the operating service environments, could be sufficient to ensure that qualification is maintained.

DISCUSSION OF TECHNICAL ASSESSMENT

The technical assessment of GSI-168 is based on reviews and analyses of the research results of six LOCA tests, condition-monitoring tests on I&C cables, and information provided by the nuclear industry. Summaries of significant research findings are presented below. Details of the NRC technical assessment of GSI-168 are available in the NRC Agencywide Documents Access and Management System (ADAMS), Accession No. ML021790551.

Current EQ Process (40 Years)

The current EQ process is adequate for assuring that low-voltage I&C cables will perform their intended functions for 40 years. When I&C cables are qualified in accordance with NRC regulations, the overall EQ process provides reasonable assurance that I&C cables will perform their intended safety-related functions during their qualified life. Specifically, 10 CFR 50.49(e) requires consideration of all significant types of aging degradation that can affect the component's functional capability. Compliance with 10 CFR 50.49 provides reasonable assurance that the cables will perform their intended functions during and following design basis events after exposure to the effects of service condition aging. Further, some licensees have implemented monitoring programs to ensure that service conditions will not exceed those assumed during the original qualification. Inspection, surveillance, condition monitoring, and trending of selected parameters for any installed safety-related cable system could increase the confidence in cable performance.

EQ Process for License Renewal (60 Years)

Licensees that have addressed license renewal recognize that knowledge of the operating service environments is essential to extending the qualified life of I&C cables. Where measured environmental service conditions are less severe than those used in the original qualification and when the cables are not degraded, the licensees assessed the difference between the operating environment and the original qualification environment to extend the qualified life of the cables to 60 years by reanalysis. This approach, based on the Arrhenius methodology, has been found acceptable by the staff during its review of license renewal applications.

Results of Cable LOCA Tests

Detailed information on the six cable LOCA tests conducted at Wyle Laboratories is provided in NUREG/CR-6704, "Assessment of Environmental Qualification Practices and Condition Monitoring Techniques for Low-Voltage Electric Cables." It should be noted that the LOCA conditions selected for the simulated tests were consistent with those used in the original qualification of the cables. All cable specimens in Test Sequences 1, 2, and 3 passed the LOCA test and the voltage withstand test. Samuel Moore cable specimens failed the voltage withstand test during Test Sequence 4, and Okonite bonded- jacket cable specimens failed the

LOCA test and the voltage withstand test in Test Sequence 5. All of the Test Sequence 6 cable specimens, aged to 60 years, exhibited high leakage currents and several cable specimens failed the voltage withstand test. The summary results of the six test sequences are discussed in Attachment 1.

Research Findings on Cable Condition-Monitoring Techniques

NRC research results on I&C cables indicate that meaningful information can be derived from testing samples of polymeric materials under controlled laboratory conditions. With certain limitations (accessibility being the biggest limitation), some of these test results can be applied in the in situ assessment of installed cable systems. The research concluded that a combination of condition-monitoring techniques could be effective since no single technique is currently adequate to detect insulation degradation of I&C cables. Based on the test results, conclusions were drawn regarding the effectiveness of the techniques studied for monitoring cable condition and are presented in the attachment.

Industry Good Practices for Condition-Monitoring

During the NRC review of GSI-168, the industry stated that cable aging evaluations are ongoing throughout plant life. When unexpected localized adverse conditions are identified, the condition of the affected cables is evaluated and appropriate corrective action is taken. Monitoring or inspection of environmental conditions or component parameters was generally conducted to ensure that the component is within the bounds of its qualification basis. The combination of licensee-specific activities and industry-supported activities that were developed for condition-monitoring can support a high level of confidence that installed safety-related cables would remain qualified to perform their safety functions in the event of an accident. In addition, the nuclear industry continues to advance the state-of-the-art in cable condition-monitoring from the simplest techniques to the most sophisticated. The staff has concluded that, although a single reliable condition-monitoring technique does not currently exist, walkdowns to look for any visible signs of anomalies attributable to cable aging, coupled with monitoring of operating environments, have proven to be effective and useful.

Risk Assessment

The state-of-the-art for incorporating cable aging effects into probabilistic risk assessment is still evolving and current assumptions that need to be made on the failure rate and common cause effects are based on sparse data. One of the key assumptions of the risk assessment is that operating environments are less severe than or the same as those assumed during qualification testing. These assumptions can be relied upon provided licensees have ongoing knowledge of environmental operating conditions at the nuclear power plants.

SUMMARY OF ISSUE

The technical assessment of GSI-168 is complete and the research findings are published in NUREG/CR-6704, Vols. 1 and 2 (Accession Nos. ML010460247 and ML010510387). The significant research findings that resulted from this effort are as follows:

- The current equipment qualification process for low-voltage I&C cables is adequate for the duration of the current license term of 40 years.
- Because of the failures of some I&C cables in the NRC LOCA tests, the original margin and conservatism inherent in the qualification process have been reduced. Adequate margin may be ensured through ongoing monitoring of plant operating environments to confirm that service conditions do not exceed those assumed during qualification testing and the cables are within the bounds of their qualification basis.
- Walkdowns, with particular emphasis on the identification of localized adverse environments, to look for any visible signs of anomalies attributable to cable aging, coupled with the monitoring of operating environments, were proven to be effective and useful for ensuring qualification of cables.
- For license renewal, a reanalysis (based on the Arrhenius methodology) to extend the life of the cables by using the available margin based on a knowledge of the actual operating environment compared to the qualification environment, coupled with observations of the condition of the cables during walkdowns, was found to be an acceptable approach.
- A combination of condition-monitoring techniques may be needed since no single technique is currently demonstrated to be adequate to detect and locate degradation of I&C cables. Monitoring I&C cable condition could provide the basis for extending cable life.

BACKFIT DISCUSSION

This RIS requests no action or written response. Consequently, the staff did not perform a backfit analysis.

***FEDERAL REGISTER* NOTIFICATION**

A notice of opportunity for public comment was not published in the *Federal Register* because this RIS is informational.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not request any information collection.

If there are any questions concerning this RIS, please contact the person noted below.

/RA/
William D. Beckner, Program Director
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical Contact: T. Koshy, NRR
301-415-1176
E-mail: txk@nrc.gov

Attachments:
1. Results of Cable LOCA Tests and
   Findings On Cable Condition-Monitoring Techniques
2. List of Recently Issued NRC Regulatory Issue Summaries

RESULTS OF CABLE LOCA TESTS AND FINDINGS ON CABLE CONDITION-MONITORING TECHNIQUES

CABLE LOCA TESTS

Detailed information on the six cable LOCA tests conducted at Wyle Laboratories is provided in NUREG/CR-6704, "Assessment of Environmental Qualification Practices and Condition Monitoring Techniques for Low-Voltage Electric Cables." It should be noted that the LOCA conditions selected for the simulated tests were consistent with those used in the original qualification of the cables. The summary results of the six test sequences are presented below.

Test Sequence 1: XLPE Insulated Cables Aged to 20 Years

The samples tested in this sequence were #14 and #16 American wire gauge (AWG) XLPE-insulated cables with a Neoprene overall outer jacket manufactured by Rockbestos, with the trade name "Firewall III." The preaging parameters for the four groups of specimens in this test sequence were as follows:

Group 1: No accelerated aging (control specimens)
Group 2: Accelerated aging to match naturally aged cable (2.86 hr @ 248 °F + 0.63 Mrad)
Group 3: Naturally aged cable (10 years old)
Group 4: Accelerated aging to 20 years (648.5 hr @ 302 °F + 26.1 Mrad)

The LOCA conditions simulated included exposure to 150 Mrad of accident radiation, followed by exposure to steam at high temperature and pressure (346 °F and 113 psig peak conditions, double-peak profile) and chemical spray. The test duration was 7 days. All cable specimens passed the LOCA test sequence, including the post-LOCA voltage withstand test.

Test Sequence 2. EPR-Insulated Cables Aged to 20 Years

The samples used in this sequence were three-conductor (3/C) and four-conductor (4/C) #16 AWG, 600v AIW cables with ethylene propylene (EPR) and unbonded chlorosulfonated polyethylene (CSPE, with the trade name Hypalon), covering the insulation of each conductor and the conductor bundle. The preaging parameters for the four groups of specimens in this test sequence were as follows:

Group 1: No accelerated aging (control specimens)
Group 2: Accelerated aging to match naturally aged cable (28.5 hr @ 250 °F + 3.3 Mrad)
Group 3: Naturally aged cable (24 years old)
Group 4: Accelerated aging to 20 years (82.2 hr @ 250 °F + 25.7 Mrad)

The LOCA conditions simulated included exposure to 150 Mrad of radiation followed by exposure to steam (340 °F and 60 psig peak conditions, single-peak profile) and chemical spray. The test duration was 7 days. All cable specimens passed the LOCA test sequence, including the post-LOCA voltage withstand test.

Test Sequence 3. XLPE-Insulated Cables Aged to 40 Years

The test specimens were cross-linked-polyethylene (XLPE)-insulated cables with a Neoprene overall outer jacket manufactured by Rockbestos, with the trade name "Firewall III." The preaging parameters for the four groups of specimens in this test sequence were as follows:

Group 1. No accelerated aging (control specimens)
Group 2. Accelerated aging to simulate the exposure of the naturally aged specimens (9.93 hr @ 248 °F + 2.27 Mrad)
Group 3. Naturally aged 10-year-old cable
Group 4. Accelerated aging to simulate 40 years of qualified life (1301.16 hr @ 302 °F + 51.49 Mrad)

The LOCA conditions simulated included exposure to 150 Mrad of accident radiation followed by exposure to steam (using the same LOCA profile as used in Test Sequence 1) and chemical spray.

One of the Group 4 specimens did not hold the full 500 volts used for insulation resistance (IR) testing even after its splices were removed. The cause of this failure was determined to be human error in handling the test specimen. With the exception of the damaged specimen, all cable specimens passed the LOCA test sequence, including the post-LOCA voltage withstand test.

Test Sequence 4. Multiconductor Cables

The objective of this test sequence was to determine whether multiconductor cables have any unique failure mechanisms that are not present in single-conductor cables. The test specimens were #12 AWG, 3/C, 1,000V EPR-insulated cables with individual and outer CSPE jackets manufactured by Anaconda. In addition, this test sequence included #16 AWG, 2/C, 600V Samuel Moore cables with ethylene propylene diene monomer (EPDM) insulation and a CSPE bonded individual jacket with a Dekorad overall outer jacket. The preaging groups in this test sequence were as follows:

Group 1. Anaconda and Samuel Moore cables with no accelerated aging (control specimens)
Group 2. Samuel Moore cables with accelerated aging to simulate 20 years of qualified life (84.85 hr @ 250 °F + 25.99 Mrad)
Group 3. Anaconda cables (169.20 hr @ 302 °F + 53.60 Mrad) and Samuel Moore cables (169.05 hr @ 250 °F + 51.57 Mrad) with accelerated aging to simulate 40 years of qualified life.

The LOCA conditions simulated included exposure to 150 Mrad of accident radiation followed by steam (346 °F and 113 psig peak conditions, as used in Test Sequences 1 and 3) and chemical spray. During the post-LOCA voltage withstand test, all of the Anaconda cables and Samuel Moore cables aged to simulate 20 years performed acceptably. However, two out of three Samuel Moore specimens aged to simulate 40 years could not hold the 2,400V test voltage on one conductor. Inspection of the two specimens revealed a single pinhole in the insulation of each failed conductor. It was concluded that the failures were due to localized degradation of the insulation, which caused the high-potential test to puncture the insulation on the two failed conductors. There was no general degradation of the insulation along the length of the cable specimens and no unique failure mechanism was observed between the single-conductor and multiconductor cables. Therefore, based on these test results, the issue of a unique failure mechanism for multiconductor vs. single-conductor low-voltage I&C cables was not demonstrated.

Test Sequence 5. Bonded Jacket Cables

The samples used in this sequence were **A**naconda 3/C, #12AWG, 1,000V cables with EPR insulation and a CSPE jacket; **S**amuel Moore 2/C, #16 AWG, 600V cables with EPDM insulation and a CSPE jacket; and **O**konite 1/C, #12 AWG, 600V cables with EPR insulation and a CSPE jacket. The preaging groups in this test sequence were as follows:

Group 1. Specimens with no accelerated aging (control specimens)
Group 2. Specimens from A, S, and O with accelerated aging to simulate 20 years of qualified life (A: 84 hr @ 302 °F + 25.69 Mrad; S: 84 hr @ 250 °F + 25.99Mrad; and O: 252 hr @ 302 °F + 25.79 Mrad)
Group 3. Specimens from A, S, and O with accelerated aging to simulate 40 years of qualified life (A: 169 hr @ 302 °F + 51.35 Mrad; S: 169 hr @ 250 °F + 51.57 Mrad; and O: 504 hr @ 302 °F + 51.49 Mrad)

The LOCA conditions simulated included exposures to 150 Mrad of accident radiation, followed by steam (double-peak LOCA profile, as used in Test Sequences 1 and 3 with a test duration of 10 days) and chemical spray. After post-LOCA inspections, a voltage withstand test was conducted on each of the cable specimens. All of the Samuel Moore and Anaconda cables performed acceptably, while one of the two Okonite specimens in Group 2 and all 3 Okonite specimens in Group 3 failed the 2,400V voltage withstand test. It was observed that the insulation on the Okonite cables had split open along their length during the simulated LOCA, exposing the bare conductor underneath. It was concluded that the failures in the Okonite specimens were caused by differential swelling of the bonded CSPE individual jacket and the underlying EPR insulation.

The Okonite Company has subsequently requalified the 1/C, #12 AWG Okonite Okolon composite cable based on an Arrhenius activation energy of 1.24eV. Calculations using this activation energy (225 hr @ 150 °C + 200 Mrad and 300 hr @ 150 °C + 100 Mrad) extrapolate to a 40-year qualified life at 75 °C and 77 °C, respectively. Additional details of the recent Okonite cable requalification program are contained in Regulatory Issue Summary 2002-11 (ADAMS Accession No. ML022190099), issued August 9, 2002.

Test Sequence 6: EPR- and XLPE-Insulated Cables Aged to 60 Years

The test specimens were Rockbestos cables (same as Test Sequences 1and 3), AIW cables (same as Test Ssequence 2), Samuel Moore cables (same as Test Sequences 4 and 5), and Okonite cables (same as Test Sequence 5). The preaging groups in this test sequence were as follows:

Group 1: No accelerated aging (control specimens)

Group 2: Rockbestos cables (1366 hr @ 302 °F +77 Mrad), Okonite cables (756 hr @ 302 °F + 77 Mrads), AIW cables (252 hr @ 250 °F + 38 Mrad), and Samuel Moore cables (252 hr @ 250 °F + 77 Mrad) with accelerated aging to simulate 60 years of qualified life.

The LOCA conditions simulated included exposure to either 75 Mrad (AIW cables only) or 150 Mrad of accident radiation, followed by exposure to steam (double-peak LOCA profile, as used in Test Sequences 1 and 3, with peak conditions of 346 °F and 113 psig and a duration of 10 days) and chemical spray.

Following the post-LOCA investigation, the test specimens were subjected to a voltage withstand test. In general, *all* of the specimens aged to 60 years exhibited a weakening of the insulation, which was manifested in the form of high leakage currents. Some of the specimens were unable to hold the required 2,400V of the voltage withstand test.

Error in Irradiation Dose

Following the completion of cable LOCA testing at Wyle Laboratories, the Georgia Institute of Technology notified Wyle Laboratories of an error in irradiation dose that affected LOCA tests 2 through 6. All specimens received irradiations from 6% to 10.5% lower than previously reported. Prior to completion of the GSI-168 technical assessment, the reported error in irradiation dose was evaluated by the Brookhaven National Laboratory and the NRC staff to determine if this error would impact the research findings. The staff's review concluded that none of the conclusions of the GSI-168 technical assessment are impacted by this error. The staff recognizes that the radiation dose of 50 Mrad used for qualification is conservative when compared to the 40-year dose seen during normal service in a nuclear power plant.

RESEARCH FINDINGS ON CABLE CONDITION-MONITORING TECHNIQUES

Based on the results of the testing, the following conclusions were drawn regarding the effectiveness of the techniques studied for monitoring cable condition.

Visual Inspection

Visual inspection does not provide quantitative data; however, it does provide useful information on the condition of the cable that is relatively easy and inexpensive to obtain and that can be used to determine whether further investigation of the cable condition is warranted. Visual inspection is demonstrated to be a valuable source of information in any cable condition-monitoring program.

Elongation at Break (EAB)

EAB was found to be a reliable technique for determining the condition of the polymers studied. While EAB provides trendable data that can be readily correlated with material condition, it is a destructive test and cannot be used as an in situ means of monitoring electric cables unless sacrificial cable specimens are available.

Oxidation Induction Time Method (OITM)

OITM was found to be a promising technique for monitoring the condition of electric cables. Results show that aging degradation can be trended with this technique for both XLPE and EPR insulation. However, a small sample of cable material is needed to perform this test.

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OIT, which is related to OITM, was found to be less sensitive for detecting aging degradation of the polymers studied. OITM is preferred at this time.

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In terms of ability to trend aging degradation in the polymers studied, FTIR spectroscopy was found to provide inconclusive results. The results tend to show a consistent trend with age. However, the technical basis for the trend remains questionable.

Indenter

The indenter was found to be a reliable device that provides reproducible, trendable data for monitoring the degradation of cables in situ. It is limited to accessible sections of the cable, but it was found to be effective for monitoring the condition of common cable jacket and insulation materials and can be used for monitoring localized and accessible segments of low-voltage electric cables.

Hardness

The results of the hardness test indicate that, over a limited range, hardness can be used to trend cable degradation. However, different probes must be used to accommodate the change in material hardness. Also, puncturing the cable insulating material is a potential concern with this technique and must be taken into consideration.

Insulation Resistance

Degradation of cable insulation can be trended with this technique. As cables degrade, a definite change in insulation resistance can be detected that can be correlated to cable condition. Using 1-minute and 10-minute readings to calculate the polarization index enables the effects of temperature and humidity variations to be accounted for. This technique can be used as an in situ condition-monitoring technique.

Dielectric Loss

This technique was found to provide useful data for trending the degradation of cable insulation. As the cables degrade, a definite change in phase angle between an applied test voltage and the circuit current can be detected at various test frequencies and correlated to cable condition. This technique can be used as an in situ condition monitoring technique. However, it is more effective when a ground plane is an integral element of a cable system.

Functional Performance

This technique alone does not provide sufficient data to determine the condition of a cable. It is a "go—no go" type of test and may not be effective in detecting degraded conditions and impending failures. Further, functional performance testing is not considered an effective method for determining, in situ, the LOCA survivabiltiy for a particular cable.

Voltage Withstand

The capability of the insulating materials to withstand the circuit voltage is an indication of its dielectric performance. In order to detect defects in an incipient state, applied voltages may have to be elevated considerably above the rated voltages of the systems; further, the equipment at both ends of a cable system under test must be either disconnected or protected. Voltage withstand tests may result in unanticipated degradation of cables and can result in failures. Therefore, the risk of causing either catastrophic or incipient damage to cable insulation makes this an unsuitable method for assessing the LOCA survivability of low-voltage electric cables in situ.

EXHIBIT DD

Department of Energy

Idaho Operations Office
1955 Fremont Avenue
Idaho Falls, ID 83415

June 29, 2007

Sherwood Martinelli
351 Dyckman Street
Peekskill, New York 10566

SUBJECT: Freedom of Information Act Request 07-044 Martinelli (OM-PA-07-060)

Dear Mr. Martinelli:

This letter is in partial response to your Freedom of Information Act (FOIA) request in which you asked the Department of Energy (DOE) for documents that relate to or are created to assist the nuclear industry to identify long-term aging issues and problems of nuclear power plants.

Your request was modified in a letter dated May 29, 2007 in which you specifically asked for the following information:

1. Correspondence from NEPO that relates to research and /or effort to address licensing/re-licensing issues of nuclear facilities from 1999 to 2006, specifically reactors, and research conducted into ageing, degradation, and security issues at nuclear facilities; and
2. Correspondence between the DOE, NRC, and NEI and/or owners of nuclear facilities from 1999 to 2006 that relates to research and/or efforts to address licensing/re-licensing issues of nuclear facilities, specifically reactors, and research conducted into ageing, degradation, and security issues at nuclear facilities.

Per a letter to you dated May 30, 2007 from Abel Lopez, Director of the FOIA/ Privacy Act Office at DOE Headquarters, your request was transferred to the DOE Office of Nuclear Energy (DOE-NE) to conduct a search of its files for any documents responsive to your request. Your request to receive a waiver of fees for processing the request was also granted at that time.

The DOE-NE has conducted a thorough search of its files located both at the DOE Headquarters Office as well as the Idaho Operations Office and has found documents responsive to your request.

Enclosed, please find 18 reports that are considered responsive to Item 1 of your request. These reports have been reviewed and determined to be fully releasable to you under the provisions of the FOIA. A complete document index is also enclosed for your convenience.
Further, several additional documents were located that may be responsive to Items 1 and 2 of your request. Due to the volume of these documents and the need to coordinate with other DOE Offices, additional time is needed to conduct a review and release determination in accordance

with the FOIA. It is estimated that a final response will be provided to you on or before October 27, 2007 regarding these additional documents.

As the Authorizing and Denying Official, it is my duty to inform you of your right to appeal this decision to the Office of Hearings and Appeals. All of the information you need to file an appeal is enclosed.

If you have any questions regarding this letter or the enclosures, please feel free to contact me at (208) 526-0709.

Sincerely,

Nicole Brooks

Nicole Brooks
Freedom of Information Officer, DOE-ID

Enclosures

EXHIBIT FF

FC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

May 2, 2003

NRC REGULATORY ISSUE SUMMARY 2003-09 ENVIRONMENTAL QUALIFICATION OF LOW-VOLTAGE INSTRUMENTATION AND CONTROL CABLES

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform addressees of the results of the technical assessment of GSI-168, "Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables." The scope of GSI-168 is limited to safety-related, low-voltage I&C cables. This RIS requires no action or written response on the part of an addressee.

BACKGROUND

In support of the resolution of GSI-168, the NRC sponsored cable test research at Wyle Laboratories and the Brookhaven National Laboratory. The resulting NRC technical assessment was essentially based on reviews and analyses of the research results of six loss-of-coolant-accident (LOCA) cable tests, condition-monitoring tests on I&C cables, and information provided by the nuclear industry. Technical assessments were coordinated with the nuclear industry and the Institute of Electrical and Electronics Engineers.

Following the completion of the NRC research effort, the staff concluded that typical I&C cable qualification test programs include numerous conservative practices that collectively provide a high level of confidence that the installed I&C cables will perform their intended functions during and following design basis events as required by 10 CFR 50.49, "Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants." These conservative practices continue to support the current use of a single prototype during qualification testing and, therefore, a successful test provides a high level of confidence that these cables will be able to perform their safety functions during and following a design basis event. However, cable LOCA test failures that occurred during the NRC-sponsored research program indicate that in certain cases the original margin and conservatism inherent in the qualification process have been reduced. Licensees have stated in a few cases that a reduction in margin can be addressed by monitoring operating service environments (temperature, radiation, and humidity) to ensure that operating conditions do not exceed the parameters that were assumed during qualification testing. In this regard, walkdowns to look for any visible signs of anomalies attributable to aging, with particular emphasis on localized adverse environments, coupled with the knowledge of the operating service environments, could be sufficient to ensure that qualification is maintained.

DISCUSSION OF TECHNICAL ASSESSMENT

The technical assessment of GSI-168 is based on reviews and analyses of the research results of six LOCA tests, condition-monitoring tests on I&C cables, and information provided by the nuclear industry. Summaries of significant research findings are presented below. Details of the NRC technical assessment of GSI-168 are available in the NRC Agencywide Documents Access and Management System (ADAMS), Accession No. ML021790551.

Current EQ Process (40 Years)

The current EQ process is adequate for assuring that low-voltage I&C cables will perform their intended functions for 40 years. When I&C cables are qualified in accordance with NRC regulations, the overall EQ process provides reasonable assurance that I&C cables will perform their intended safety-related functions during their qualified life. Specifically, 10 CFR 50.49(e) requires consideration of all significant types of aging degradation that can affect the component's functional capability. Compliance with 10 CFR 50.49 provides reasonable assurance that the cables will perform their intended functions during and following design basis events after exposure to the effects of service condition aging. Further, some licensees have implemented monitoring programs to ensure that service conditions will not exceed those assumed during the original qualification. Inspection, surveillance, condition monitoring, and trending of selected parameters for any installed safety-related cable system could increase the confidence in cable performance.

EQ Process for License Renewal (60 Years)

Licensees that have addressed license renewal recognize that knowledge of the operating service environments is essential to extending the qualified life of I&C cables. Where measured environmental service conditions are less severe than those used in the original qualification and when the cables are not degraded, the licensees assessed the difference between the operating environment and the original qualification environment to extend the qualified life of the cables to 60 years by reanalysis. This approach, based on the Arrhenius methodology, has been found acceptable by the staff during its review of license renewal applications.

Results of Cable LOCA Tests

Detailed information on the six cable LOCA tests conducted at Wyle Laboratories is provided in NUREG/CR-6704, "Assessment of Environmental Qualification Practices and Condition Monitoring Techniques for Low-Voltage Electric Cables." It should be noted that the LOCA conditions selected for the simulated tests were consistent with those used in the original qualification of the cables. All cable specimens in Test Sequences 1, 2, and 3 passed the LOCA test and the voltage withstand test. Samuel Moore cable specimens failed the voltage withstand test during Test Sequence 4, and Okonite bonded- jacket cable specimens failed the

LOCA test and the voltage withstand test in Test Sequence 5. All of the Test Sequence 6 cable specimens, aged to 60 years, exhibited high leakage currents and several cable specimens failed the voltage withstand test. The summary results of the six test sequences are discussed in Attachment 1.

Research Findings on Cable Condition-Monitoring Techniques

NRC research results on I&C cables indicate that meaningful information can be derived from testing samples of polymeric materials under controlled laboratory conditions. With certain limitations (accessibility being the biggest limitation), some of these test results can be applied in the in situ assessment of installed cable systems. The research concluded that a combination of condition-monitoring techniques could be effective since no single technique is currently adequate to detect insulation degradation of I&C cables. Based on the test results, conclusions were drawn regarding the effectiveness of the techniques studied for monitoring cable condition and are presented in the attachment.

Industry Good Practices for Condition-Monitoring

During the NRC review of GSI-168, the industry stated that cable aging evaluations are ongoing throughout plant life. When unexpected localized adverse conditions are identified, the condition of the affected cables is evaluated and appropriate corrective action is taken. Monitoring or inspection of environmental conditions or component parameters was generally conducted to ensure that the component is within the bounds of its qualification basis. The combination of licensee-specific activities and industry-supported activities that were developed for condition-monitoring can support a high level of confidence that installed safety-related cables would remain qualified to perform their safety functions in the event of an accident. In addition, the nuclear industry continues to advance the state-of-the-art in cable condition-monitoring from the simplest techniques to the most sophisticated. The staff has concluded that, although a single reliable condition-monitoring technique does not currently exist, walkdowns to look for any visible signs of anomalies attributable to cable aging, coupled with monitoring of operating environments, have proven to be effective and useful.

Risk Assessment

The state-of-the-art for incorporating cable aging effects into probabilistic risk assessment is still evolving and current assumptions that need to be made on the failure rate and common cause effects are based on sparse data. One of the key assumptions of the risk assessment is that operating environments are less severe than or the same as those assumed during qualification testing. These assumptions can be relied upon provided licensees have ongoing knowledge of environmental operating conditions at the nuclear power plants.

SUMMARY OF ISSUE

The technical assessment of GSI-168 is complete and the research findings are published in NUREG/CR-6704, Vols. 1 and 2 (Accession Nos. ML010460247 and ML010510387). The significant research findings that resulted from this effort are as follows:

- The current equipment qualification process for low-voltage I&C cables is adequate for the duration of the current license term of 40 years.

- Because of the failures of some I&C cables in the NRC LOCA tests, the original margin and conservatism inherent in the qualification process have been reduced. Adequate margin may be ensured through ongoing monitoring of plant operating environments to confirm that service conditions do not exceed those assumed during qualification testing and the cables are within the bounds of their qualification basis.

- Walkdowns, with particular emphasis on the identification of localized adverse environments, to look for any visible signs of anomalies attributable to cable aging, coupled with the monitoring of operating environments, were proven to be effective and useful for ensuring qualification of cables.

- For license renewal, a reanalysis (based on the Arrhenius methodology) to extend the life of the cables by using the available margin based on a knowledge of the actual operating environment compared to the qualification environment, coupled with observations of the condition of the cables during walkdowns, was found to be an acceptable approach.

- A combination of condition-monitoring techniques may be needed since no single technique is currently demonstrated to be adequate to detect and locate degradation of I&C cables. Monitoring I&C cable condition could provide the basis for extending cable life.

BACKFIT DISCUSSION

This RIS requests no action or written response. Consequently, the staff did not perform a backfit analysis.

***FEDERAL REGISTER* NOTIFICATION**

A notice of opportunity for public comment was not published in the *Federal Register* because this RIS is informational.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not request any information collection.

If there are any questions concerning this RIS, please contact the person noted below.

/RA/
William D. Beckner, Program Director
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical Contact: T. Koshy, NRR
301-415-1176
E-mail: txk@nrc.gov

Attachments:
1. Results of Cable LOCA Tests and
   Findings On Cable Condition-Monitoring Techniques
2. List of Recently Issued NRC Regulatory Issue Summaries

Test Sequence 3. XLPE-Insulated Cables Aged to 40 Years

The test specimens were cross-linked-polyethylene (XLPE)-insulated cables with a Neoprene overall outer jacket manufactured by Rockbestos, with the trade name "Firewall III." The preaging parameters for the four groups of specimens in this test sequence were as follows:

Group 1. No accelerated aging (control specimens)
Group 2. Accelerated aging to simulate the exposure of the naturally aged specimens (9.93 hr @ 248 °F + 2.27 Mrad)
Group 3. Naturally aged 10-year-old cable
Group 4. Accelerated aging to simulate 40 years of qualified life (1301.16 hr @ 302 °F + 51.49 Mrad)

The LOCA conditions simulated included exposure to 150 Mrad of accident radiation followed by exposure to steam (using the same LOCA profile as used in Test Sequence 1) and chemical spray.

One of the Group 4 specimens did not hold the full 500 volts used for insulation resistance (IR) testing even after its splices were removed. The cause of this failure was determined to be human error in handling the test specimen. With the exception of the damaged specimen, all cable specimens passed the LOCA test sequence, including the post-LOCA voltage withstand test.

Test Sequence 4. Multiconductor Cables

The objective of this test sequence was to determine whether multiconductor cables have any unique failure mechanisms that are not present in single-conductor cables. The test specimens were #12 AWG, 3/C, 1,000V EPR-insulated cables with individual and outer CSPE jackets manufactured by Anaconda. In addition, this test sequence included #16 AWG, 2/C, 600V Samuel Moore cables with ethylene propylene diene monomer (EPDM) insulation and a CSPE bonded individual jacket with a Dekorad overall outer jacket. The preaging groups in this test sequence were as follows:

Group 1. Anaconda and Samuel Moore cables with no accelerated aging (control specimens)
Group 2. Samuel Moore cables with accelerated aging to simulate 20 years of qualified life (84.85 hr @ 250 °F + 25.99 Mrad)
Group 3. Anaconda cables (169.20 hr @ 302 °F + 53.60 Mrad) and Samuel Moore cables (169.05 hr @ 250 °F + 51.57 Mrad) with accelerated aging to simulate 40 years of qualified life.

The LOCA conditions simulated included exposure to 150 Mrad of accident radiation followed by steam (346 °F and 113 psig peak conditions, as used in Test Sequences 1 and 3) and chemical spray. During the post-LOCA voltage withstand test, all of the Anaconda cables and Samuel Moore cables aged to simulate 20 years performed acceptably. However, two out of three Samuel Moore specimens aged to simulate 40 years could not hold the 2,400V test voltage on one conductor. Inspection of the two specimens revealed a single pinhole in the insulation of each failed conductor. It was concluded that the failures were due to localized degradation of the insulation, which caused the high-potential test to puncture the insulation on the two failed conductors. There was no general degradation of the insulation along the length of the cable specimens and no unique failure mechanism was observed between the single-conductor and multiconductor cables. Therefore, based on these test results, the issue of a unique failure mechanism for multiconductor vs. single-conductor low-voltage I&C cables was not demonstrated.

Test Sequence 5. Bonded Jacket Cables

The samples used in this sequence were **A**naconda 3/C, #12AWG, 1,000V cables with EPR insulation and a CSPE jacket; **S**amuel Moore 2/C, #16 AWG, 600V cables with EPDM insulation and a CSPE jacket; and **O**konite 1/C, #12 AWG, 600V cables with EPR insulation and a CSPE jacket. The preaging groups in this test sequence were as follows:

Group 1. Specimens with no accelerated aging (control specimens)
Group 2. Specimens from A, S, and O with accelerated aging to simulate 20 years of qualified life (A: 84 hr @ 302 °F + 25.69 Mrad; S: 84 hr @ 250 °F + 25.99Mrad; and O: 252 hr @ 302 °F + 25.79 Mrad)
Group 3. Specimens from A, S, and O with accelerated aging to simulate 40 years of qualified life (A: 169 hr @ 302 °F + 51.35 Mrad; S: 169 hr @ 250 °F + 51.57 Mrad; and O: 504 hr @ 302 °F + 51.49 Mrad)

The LOCA conditions simulated included exposures to 150 Mrad of accident radiation, followed by steam (double-peak LOCA profile, as used in Test Sequences 1 and 3 with a test duration of 10 days) and chemical spray. After post-LOCA inspections, a voltage withstand test was conducted on each of the cable specimens. All of the Samuel Moore and Anaconda cables performed acceptably, while one of the two Okonite specimens in Group 2 and all 3 Okonite specimens in Group 3 failed the 2,400V voltage withstand test. It was observed that the insulation on the Okonite cables had split open along their length during the simulated LOCA, exposing the bare conductor underneath. It was concluded that the failures in the Okonite specimens were caused by differential swelling of the bonded CSPE individual jacket and the underlying EPR insulation.

The Okonite Company has subsequently requalified the 1/C, #12 AWG Okonite Okolon composite cable based on an Arrhenius activation energy of 1.24eV. Calculations using this activation energy (225 hr @ 150 °C + 200 Mrad and 300 hr @ 150 °C + 100 Mrad) extrapolate to a 40-year qualified life at 75 °C and 77 °C, respectively. Additional details of the recent Okonite cable requalification program are contained in Regulatory Issue Summary 2002-11 (ADAMS Accession No. ML022190099), issued August 9, 2002.

Test Sequence 6: EPR- and XLPE-Insulated Cables Aged to 60 Years

The test specimens were Rockbestos cables (same as Test Sequences 1and 3), AIW cables (same as Test Ssequence 2), Samuel Moore cables (same as Test Sequences 4 and 5), and Okonite cables (same as Test Sequence 5). The preaging groups in this test sequence were as follows:

Group 1: No accelerated aging (control specimens)

Group 2: Rockbestos cables (1366 hr @ 302 °F +77 Mrad), Okonite cables (756 hr @ 302 °F + 77 Mrads), AIW cables (252 hr @ 250 °F + 38 Mrad), and Samuel Moore cables (252 hr @ 250 °F + 77 Mrad) with accelerated aging to simulate 60 years of qualified life.

The LOCA conditions simulated included exposure to either 75 Mrad (AIW cables only) or 150 Mrad of accident radiation, followed by exposure to steam (double-peak LOCA profile, as used in Test Sequences 1 and 3, with peak conditions of 346 °F and 113 psig and a duration of 10 days) and chemical spray.

Following the post-LOCA investigation, the test specimens were subjected to a voltage withstand test. In general, *all* of the specimens aged to 60 years exhibited a weakening of the insulation, which was manifested in the form of high leakage currents. Some of the specimens were unable to hold the required 2,400V of the voltage withstand test.

Error in Irradiation Dose

Following the completion of cable LOCA testing at Wyle Laboratories, the Georgia Institute of Technology notified Wyle Laboratories of an error in irradiation dose that affected LOCA tests 2 through 6. All specimens received irradiations from 6% to 10.5% lower than previously reported. Prior to completion of the GSI-168 technical assessment, the reported error in irradiation dose was evaluated by the Brookhaven National Laboratory and the NRC staff to determine if this error would impact the research findings. The staff's review concluded that none of the conclusions of the GSI-168 technical assessment are impacted by this error. The staff recognizes that the radiation dose of 50 Mrad used for qualification is conservative when compared to the 40-year dose seen during normal service in a nuclear power plant.

RESEARCH FINDINGS ON CABLE CONDITION-MONITORING TECHNIQUES

Based on the results of the testing, the following conclusions were drawn regarding the effectiveness of the techniques studied for monitoring cable condition.

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EXHIBIT JJ

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

James Comiotes
Director, Nuclear Safety Assurance
Tel 914 271 7130

July 31, 2006

Re: Indian Point Units 1, 2 and 3
Docket Nos. 50-003, 50-247 and 50-286
NL-06-079

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Stop O-P1-17
Washington, DC 20555-0001

Subject: **Ground Water Protection Baseline Information**
Indian Point Energy Center – Units 1, 2 and 3

Dear Sir or Madam:

The nuclear industry, in conjunction with the Nuclear Energy Institute (NEI), developed a questionnaire to facilitate compilation of baseline information regarding the current status of site programs for monitoring and protecting ground water. All participating nuclear sites agreed to provide the requested information to both NEI and the Nuclear Regulatory Commission.

Attachment 1 to this letter contains the questionnaire response for Indian Point Energy Center (IPEC). Please contact Mr. Patric W. Conroy at (914) 734-6668 if you have any questions or comments regarding this submittal.

There are no new commitments contained in this submittal.

Sincerely,

Patric W. Conroy for
James Comiotes
Director, Nuclear Safety Assurance
Indian Point Energy Center

Attachment 1 (Ground Water Protection Questionnaire Response)

cc: see next page

IE25

cc: Mr. John P. Boska
U.S. Nuclear Regulatory Commission

Mr. Samuel J. Collins
U.S. Nuclear Regulatory Commission

Resident Inspector's Office
Indian Point Unit 2 Nuclear Power Plant
U.S. Nuclear Regulatory Commission

Mr. Paul Eddy
New York State Dept. of Public Service

Mr. Ralph Anderson
Nuclear Energy Institute

ATTACHMENT 1 TO NL-06-079

GROUND WATER PROTECTION QUESTIONNAIRE RESPONSE

INDIAN POINT UNITS 1, 2 and 3

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 1, 2 AND 3
DOCKET NOS. 50-003, 50-247, AND 50-286

Ground Water Protection Questionnaire Response
Indian Point Energy Center (IPEC)

1. Briefly describe the program and/or methods used for detection of leakage or spills from plant systems, structures, and components that have a potential for an inadvertent release of radioactivity from plant operations into ground water.

   Response: IPEC has identified radioactive contamination in its on-site ground water. This contamination is currently being characterized to determine the sources of this contamination, as well as the nature and extent of the resulting ground water contamination plumes. As such, IPEC's ground water monitoring program is primarily focused on identifying the source of and characterizing after the fact release conditions. However, the program does include provisions for detecting leakage from potential future inadvertent releases to ground water. They include
   - Operator plant rounds include inspection for leaks and spills,
   - Radiation Protection surveys include inspection for leaks and spills,
   - Leaks/spills documented in corrective action program,
   - Inspection of systems, structures and components to identify potential leak points,
   - Radioactive Effluent Monitoring Program (REMP) Sampling,
   - Storm drain periodic sampling program, and
   - Corrective action program reporting/trending.

2. Briefly describe the program and/or methods for monitoring onsite ground water for the presence of radioactivity released from plant operations.

   Response: IPEC is in the process of investigating known Tritium and Sr-90 ground water contamination, resulting from leaks from the Unit 1 and 2 spent fuel pools (SFP). Other potential sources of leakage are also within the scope of this investigation. To accomplish this objective, a program for characterizing the nature and extent of the resulting ground water contamination and the site's hydro-geological characteristics is being conducted. As a part of this program, more than 30 monitoring wells have been installed throughout the site for the purpose of sampling ground water and obtaining hydro-geological data. These monitoring wells are sampled on a periodic basis, with the samples analyzed for Tritium, Sr-90 and gamma emitters. Upon conclusion of this investigation and any warranted remediation, these investigation monitoring wells will be transitioned into a long-term ground water monitoring program.

3. If applicable, briefly summarize any occurrences of inadvertent releases of radioactive liquids that have been documented in accordance with 10 CFR 50.75(g).

   Response: The most significant sources for potential releases to ground water include leakage from the Unit 1 and 2 SFPs, storm drains with contaminated sediment resulting from past spills, and an impoundment containing contaminated soil from a Unit 1 septic leach field that was excavated for construction of Unit 3. Other smaller inadvertent releases and spills have also occurred.

FAC

EXHIBIT KK

Indian Point 2
2Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO INCORPORATE DESIGN BASIS INFORMATION INTO PROCEDURES TO ASSURE ADEQUATE COOLING WATER FLOW TO THE RCP THERMAL BARRIERS
The inspectors identified a Green, non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not appropriately incorporate design requirements into an operating procedure used to establish adequate component cooling water (CCW) flow to the reactor coolant pump (RCP) thermal barriers. Specifically, the flow specification in the CCW operating procedure did not incorporate the calculated design flow requirements to bound allowable CCW temperature limits. Entergy entered this issue into their corrective action program and will be evaluating the flow requirements specified in procedure 2-SOP-4.1.2, "Component Cooling Water System Operation," to ensure that they bound the allowed plant operating limits.

The inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, Entergy did not incorporate design flow requirements necessary to assure adequate cooling water flow to the RCP thermal barriers into the plant operating procedures which establish the required flow. On a loss of seal injection, the procedure did not ensure that the heat removal capability was adequate to prevent a rise in seal temperature which would require the RCP to be stopped with a subsequent reactor trip. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This finding was determined to be of very low safety significance because it would not result in exceeding the Technical Specification limit for identified reactor coolant system leakage and would not have likely affected other mitigating systems resulting in a loss of their safety function. The inspectors found that the procedurally established nominal flow band would have assured adequate cooling of the RCP thermal barriers for the highest CCW supply temperature recorded over the previous year.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the operating procedure used to set the flow rate of cooling water to the RCP thermal barriers was not adequate to make certain that sufficient cooling water was available to assure the components could perform their design function. (Section 1R15)

Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ESTABLISH TESTING TO ASSURE ADEQUATE COOLING WATER FLOW TO THE RCP THERMAL BARRIERS
The inspectors identified a Green, NCV of 10 CFR 50 Appendix B, Criterion XI, "Test Control," in that, Entergy did not establish appropriate testing to assure adequate component cooling water (CCW) flow to the reactor coolant pump thermal barriers. Specifically no preventive maintenance activities or functional checks were conducted for the individual flow meters. It was determined that the rotameters on 21 and 23 RCP were not indicating correctly and that actual CCW flow to the thermal barrier heat exchangers was less that the design requirements for CCW temperature. Entergy entered this issue into their corrective action program (CR-IP2-2007-00783 and 00955), adjusted individual cooling water flow within the

nominal band using ultrasonic flow meters, wrote work orders to replace the faulty flow meters, and is conducting an evaluation to determine the appropriate test requirements for the flow indicators.

This inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, Entergy's test program did not assure that all testing required to demonstrate that the RCP thermal barriers will perform satisfactorily in service because no testing was performed to ensure the accuracy of the individual flow meters used to establish the required cooling water flow. Consequently, it was identified that two individual flow indicators did not read correctly and the CCW flow to two RCP's was not sufficient to assure adequate cooling in the event that seal water was lost based on the flow requirements established in design calculations. On a loss of seal injection, the cooling water flow would not ensure that the heat removal capability was adequate to prevent a rise in seal temperature which would require the RCP to be stopped with a subsequent reactor trip. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This finding was determined to be of very low safety significance because it would not result in exceeding the Technical Specification limit for identified reactor coolant system leakage and would not have likely affected other mitigating systems resulting in a loss of their safety function. (Section 1R15)

Inspection Report# : 2007002 *(pdf)*

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE RISK ASSESSMENT FOR 21 MBFP STEAM INLET VALVE
The inspectors identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (CFR), Part 50.65(a)(4), because Entergy did not adequately assess and manage the risk of on-line maintenance activities while operating with a degraded steam inlet valve on one of Entergy's two main boiler feed pumps (MBFP). Specifically, from November 16 through 21, 2006, the degraded condition of the 21 MBFP increased the likelihood of a reactor trip, but was not assessed or included in the plant's on-line risk model. Entergy entered this issue into their corrective action program and properly assessed 21 MBFP risk on November 21, 2006.

The inspectors determined that this finding was more than minor because Entergy failed to consider risk significant structures, systems, components, and support systems that were unavailable during the performance of on-line maintenance. Specifically, Entergy failed to assess the increase in online risk from the increased likelihood of a reactor trip due to the 21 MBFP degraded condition. The inspectors evaluated this finding using IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," and determined that this finding was of very low safety significance because the finding resulted in an increase in the incremental core damage probability of less than 1x10-6 (actual increase was approximately 2x10-8).

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not provide complete and accurate procedures, in that, the online risk assessment procedure did not require degraded equipment that impacted risk to be assessed or managed.
Inspection Report# : 2006005 *(pdf)*

Significance: G Sep 30, 2006
Identified By: Self-Revealing
Item Type: FIN Finding
INADEQUATE OPERATING PROCEDURES FOR LOSS OF BOTH HEATER DRAIN TANK PUMPS
A Green self-revealing finding was identified because Entergy failed to develop adequate procedures for governing the response to a loss of both heater drain tank pumps and to an approaching rod insertion limit (RIL) alarm condition. Specifically, the procedure governing operator actions during a loss of heater drain tank pumps did not specify for the operators to reset the steam dumps following the rapid downpower. The alarm response procedure for the approaching rod insertion limit condition directed the operators to place the rod control system in manual to stop further automatic inward rod motion. This impacted operators ability to add negative reactivity and control the transient. Entergy entered these procedural deficiencies into their corrective action program and is evaluating the appropriate steps to correct the procedural deficiencies.

The inspectors determined that this finding is greater than minor because it is associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, the procedural inadequacies complicated operator actions to a rapid downpower, resulted in a manual reactor trip when the operators determined that they did not have sufficient control of the transient, and could impact other accident sequences requiring negative reactivity addition. The inspectors evaluated this finding using Phase I of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined it to be of very low safety significance because it did not

contribute to the likelihood of both a reactor trip and the likelihood that mitigation equipment or functions would be unavailable. The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that plant operating procedures were adequate to ensure operators could appropriately respond to a rapid downpower transient.
Inspection Report# : 2006004 *(pdf)*

Significance: G Sep 30, 2006
Identified By: Self-Revealing
Item Type: FIN Finding
INADEQUATE PROCEDURE FOR CALIBRATING THE STEAM DUMP LOSS OF LOAD CONTROLLER
A Green self-revealing finding was identified because Entergy failed to develop an accurate procedure for calibration of the steam dump loss of load controller. This resulted in the steam dumps failing to operate properly during a plant transient, complicating operator response, and leading to a manual reactor trip. Following identification of the issue, Entergy entered the issue into the corrective action program, corrected the procedural deficiency, and re-calibrated the controller.

The inspectors determined that this finding is greater than minor because it is associated with the Procedural Quality attribute of the Initiating events cornerstone; and, it impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, the inadequacy in Entergy's calibration procedure caused the steam dumps to operate improperly during a plant transient and contributed to a reactor trip. The inspectors evaluated this finding using Phase I of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined it to be of very low safety significance because it did not contribute to the likelihood of both a reactor trip and the likelihood that mitigation equipment or functions would be available. The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that the procedure for calibration of the steam dump loss of load controller was accurate, in that, it specified incorrect settings for the controller.
Inspection Report# : 2006004 *(pdf)*

Mitigating Systems

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE DESIGN CONTROL ASSOCIATED WITH VORTEXING AND NET POSITIVE SUCTION HEAD CALCULATIONS
The team identified a finding of very low significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not ensure adequate suction submergence for the three safety injection (SI) pumps by not properly translating vortex and net positive suction head (NPSH) design parameters into calculations relative to reactor water storage tank (RWST) level. Specifically, Entergy used a non-conservative method to calculate the level required to prevent pump vortexing, and used a non-conservative RWST level value for determining available NPSH for the SI pumps. Entergy entered the issue into their corrective action program and revised the affected calculations.

The finding is more than minor because the calculation deficiencies represented reasonable doubt on the operability of the SI pumps, even though the pumps were ultimately shown to be operable. The finding is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the significance determination process (SDP), documented in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because it was a design deficiency that did not result in a loss of SI system operability, based upon the team's verification of Entergy's revised calculations.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE DIFFERENTIAL PRESSURE VALUE USED FOR MOV 746 AND MOV 747 TONENSURE VALVE CAPABILITY
The team identified a finding of very low significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not accurately incorporate design parameters into valve thrust calculations for motor operated valve (MOV) 746 and MOV 747. Specifically, Entergy used an incorrect and non-conservative differential pressure in the calculations for MOV 746 and MOV 747, which were developed to verify that the valves could develop sufficient thrust to open under postulated design basis conditions. Additionally, an incorrect equation was used in determining the reduction

in motor torque due to degraded voltage conditions. Entergy entered the issue into their corrective action program and revised the affected calculations using the correct information.

The finding is more than minor because the calculation deficiencies represented reasonable doubt on the operability of MOV 746 and MOV 747. The finding is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it was a design deficiency that did not to result in a loss of MOV 746 and MOV 747 operability, based upon the team's verification of Entergy's revised calculations.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE DESIGN CONTROL FOR ENVIRONMENTAL EFFECTS TO ENSURE THE AVAILABILITY OF THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP OPERATION
The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not establish adequate design control measures to ensure the availability of the turbine driven auxiliary feedwater pump (TDAFWP) during a postulated loss-of-offsite power (LOOP) event. Under certain LOOP situations, the team determined that the TDAFWP steam supply could be inadvertently isolated because of inadequate calculations and procedures for limiting the AFWP room temperature rise. Specifically, a calculation to determine the auxiliary feedwater pump (AFWP) room temperature rise during a LOOP did not include heat input from the TDAFWP. Further, actions that could limit the rise in AFWP room temperature and prevent the inadvertent isolation of the TDAFW pump (opening an AFWP room roll-up door or promptly restoring forced ventilation) were not included in procedures. Entergy entered this issue into their corrective action program, implemented immediate compensatory actions, and revised AFWP room temperature rise calculations.

The finding is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of safety function of the TDAFWP (single train) for greater than its 72 hour technical specification allowed outage time, based on the team's review and assessment of site ambient temperature data over the last year.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ADEQUATELY MONITOR GAS TURBINE SYSTEM PERFORMANCE AS REQUIRED BY THE MAINTENANCE RULE
The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50.65(a)(1), the Maintenance Rule, in that, Entergy failed to monitor the gas turbine (GT) system in a manner that provided reasonable assurance that the system could perform its intended safety function. Specifically, Entergy did not establish appropriate GT reliability goals, and therefore did not take corrective actions, when GT-1 had exceeded these goals for maintenance preventable functions failures (MPFF). In addition, Entergy did not properly classify repeat MPFFs, which resulted in a similar failure to take corrective actions as required. This resulted in additional GT-1 out of service time that would not have happened if appropriate actions had been taken. Entergy entered this issue into their corrective action program and lowered the allowable goal for MPFFs, and revised the GT-1 (a)(1) action plan to improve reliability.

The finding is more than minor because appropriate GT reliability goals were not established commensurate with safety and appropriate corrective actions were not taken when goals were not met. This finding is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 and Phase 2 of the SDP, which considered that the additional GT-1 out of service time due to this issue could be as much as three days. The finding has a cross-cutting aspect in the area of human performance because Entergy did not adequately ensure procedures were complete, accurate, and up-to-date. Specifically, procedure ENN-DC-171, "Maintenance Rule Monitoring," did not provide steps to discriminate between the classification of an initial design deficiency and further failures due to the same condition, resulting in mis-classifying several GT functional failures.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC

Item Type: FIN Finding
FAILURE TO CORRECT DEGRADED GAS TURBINE 1 RELIABILITY
The team identified a finding of very low safety significance involving Entergy procedure, EN-LI-102, "Corrective Action Process," in that, Entergy failed to take corrective actions to address degraded GT-1 reliability. This resulted in a two and one half day time period in January 2007 when GT-1 and GT-3 were simultaneously inoperable because, after GT-3 was made inoperable for planned maintenance activities, GT-1 was subsequently found to be inoperable. Specifically, the reliability of GT-1 declined from an average of 75% for 2005 and the first 10 months of 2006, to 50% for the three months from November 2006 to January 2007; however, Entergy did not take actions to correct this degraded reliability. Entergy entered this issue into their corrective action program and developed an action plan to address GT reliability issues.

The issue is more than minor because it is associated with the equipment reliability attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 and Phase 2 of the SDP, assuming that both GT-1 and GT-3 were unavailable for the two and one half days, due to this issue. The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy did not correct degraded reliability of GT-1, resulting in having GT-1 and GT-3 simultaneously inoperable.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE STATION BATTERY CAPACITY TESTING FOR DEGRADATION MONITORING
The team identified a finding of very low safety significance (Green) involving a non-cited violation of Technical Specification 3.8.6.6, in that, Entergy did not perform station battery capacity testing in accordance with IEEE Standard 450-1995 (related to battery maintenance and testing). Specifically, Entergy procedurally terminated battery capacity testing at the rated discharge time (four hours), before reaching the minimum voltage, as specified by IEEE Standard 450-1995. This prevented accurate quantitative measurement of capacity degradation and identification of the need to conduct potential accelerated battery testing, as specified by both IEEE Standard 450-1995 and the technical specifications, if battery capacity drops by more than 10% relative to the previous test. Entergy entered the issue into their corrective action program and performed calculations using past test data, which demonstrated that the capacities of station batteries had not degraded more than 10%.

This issue is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of station battery safety function, based upon the team's verification of Entergy's calculations.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INEFFECTIVE CORRECTIVE ACTION FOR HIGH INTER-TIER BATTERY RESISTANCES
The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in that, Entergy did not take effective corrective actions for a condition adverse to quality concerning out-of-tolerance inter-tier resistances on the No. 21 station battery. Specifically, after repeated failures of the No. 21 station battery inter-tier resistance testing, vendor and IEEE Standard 450-1995 recommended corrective actions were not taken to correct the adverse out-of-tolerance resistance trend. Entergy entered the issue into their corrective action program and performed calculations, which demonstrated that the voltage drop due to the as-found resistance of the inter-tier connections was small and did not impact No. 21 battery operability.

This issue is more than minor because if it was left uncorrected, it would have become a more significant safety concern. Specifically, high resistance connections in a battery that is loaded during accident conditions can cause localized heating and can cause permanent damage to the battery. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of No. 21 station battery safety function, based upon the team's verification of Entergy's revised calculations. The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy did not take effective corrective actions to address the adverse trend of out-of-tolerance inter-tier resistances.
Inspection Report# : 2007007 *(pdf)*

Significance: G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation

UNTIMELY CORRECTIVE ACTIONS FOR DECREASE IN BATTERY MARGIN
The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in that, Entergy did not promptly identify and correct a condition adverse to quality, with respect to known errors in the No. 23 station battery design calculations. Specifically, Entergy did not recognize at the appropriate time the need to write a condition report, perform an operability determination, or place controls on the use of the No. 23 battery design calculations when errors were discovered in the No. 23 battery design calculations that significantly lowered the battery capacity margin. Entergy entered the issue into their corrective action program and performed calculations, which demonstrated No. 23 station battery operability through the next refueling outage, based on the calculated margin and conservatisms available.

This issue is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on Phase 1 of the SDP, because it did not represent the loss of No. 23 station battery safety function, based upon the team's verification of Entergy's revised calculations.

The finding has a cross-cutting aspect in the area of problem identification and resolution because Entergy failed to promptly identify the decrease in margin found in the No. 23 battery design calculations of record.
Inspection Report# : 2007007 *(pdf)*

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: FIN Finding
FAILURE TO IMPLEMENT CORRECTIVE ACTIONS TO CORRECT A DEGRADED CONDITION WHICH IMPACTED GAS TURBINE #1 RELIABILITY AND AVAILABILITY
The inspectors identified a Green finding, in that, Entergy's corrective actions were inadequate to resolve a deficiency associated with the gas turbine 1 (GT-1) starting diesel. This deficiency was identified following a failure of GT-1 to start on February 7, 2005, and resulted in three subsequent failures. A corrective action was written to correct the deficient condition following the initial failure and was closed on June 22, 2005, with no actions taken based on a senior management decision to cancel preventive maintenance activities on the gas turbines due to pending system retirement. Entergy entered this issue into their corrective action program and installed a modification to the coolant system to prevent further trips due to this condition.

The inspectors determined that this finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, it impacted GT-1 reliability, in that, the deficiency resulted in multiple failures to start on demand after the condition was identified and the action to correct the condition was closed without being implemented. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that a Phase 2 evaluation was required because the finding represented an actual loss of safety function of a non-Technical Specification required train of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours. The inspectors used the Risk-Informed Inspection Notebook for Indian Point Nuclear Generating Unit 2, to conduct the Phase 2 evaluation. The inspectors determined that 65 hours of unavailability were caused by the additional failures of GT-1 due to the starting diesel coolant system deficiency. The inspectors conservatively equated this cumulative unavailability time to the total exposure time and used an initiating events likelihood of less than three days. The Phase 2 approximation yielded a result of very low safety significance (Green).

The inspectors determined this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that equipment and resources were available and adequate to assure reliable operation of GT-1. Specifically, Entergy did not minimize long-standing equipment issues and maintenance deferrals associated with the gas turbine system.
Inspection Report# : 2006005 *(pdf)*

Significance: G Dec 05, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO IDENTIFY A DEGRADED CONDITION OF AN AUXILIARY FEED WATER CHECK VALVE IN THE CORRECTIVE ACTION PROGRAM
The inspectors identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in that, Entergy failed to identify a condition adverse to quality associated with improper internal clearances on BFD-68, an auxiliary feedwater check valve, in the corrective action program. Specifically, upon inspection in September 2006, the gasket between the valve's body to bonnet seal was found over-crushed causing the gasket to partially unwind, potentially impacting valve operation. Gasket damage was noted in work orders during internal valve inspections of BFD-68 performed in 1997 and 2002; however, the deficiencies were not identified in the corrective action program. Consequently, the

problem was not evaluated and corrected prior to reassembly of the valve. Entergy entered this issue into the corrective action program, evaluated the condition, and conducted repairs to the valve to ensure the proper gasket crush was obtained.

The inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding was of very low safety significance because it was not a design or qualification deficiency; it did not result in the loss of a system safety function or a train safety function for greater than the Technical Specification Allowed Outage Time; and it did not screen as potentially risk significant due to external events.
Inspection Report# : 2006006 *(pdf)*

Significance: G Dec 05, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE EVALUATION OF LEAKING 22 STEAM GENERATOR LOW FLOW BYPASS VALVE FCV-427L
A self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified, in that, Entergy failed to adequately evaluate leakage into the 22 steam generator. During the Indian Point Unit 2 reactor trip on August 23, 2006, main feedwater low flow bypass valve FCV-427L leaked excessively and resulted in an uncontrolled rise in 22 steam generator level; operator response to isolate feedwater to the steam generator in accordance with emergency operating procedures; and automatic actuation of the feedwater isolation system. The excessive leakage condition into the 22 steam generator was identified on April 4, 2006, prior to Indian Point Unit 2 refueling outage 2R17, but was not fully evaluated or corrected prior to the reactor trip on August 23, 2006. This issue was entered into the corrective action program, and FCV-427L was repaired and retested satisfactorily.

The inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of the finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding was of very low safety significance because it was not a design or qualification deficiency; it did not result in the loss of a system safety function or a train safety function for greater than the Technical Specification Allowed Outage Time; and it did not screen as potentially risk significant due to external events.

The inspectors determined that the finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not thoroughly evaluate the cause of excessive leakage into the 22 steam generator such that the resolutions addressed the causes and extent of condition of the problem.
Inspection Report# : 2006006 *(pdf)*

Barrier Integrity

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO MOVE CONTAINMENT HYDROGEN ANALYZERS TO 10 CFR 50.65 (A)(1) STATUS
The inspectors identified a Green, NCV of 10 CFR 50.65(a)(2) because Entergy did not demonstrate that the performance or condition of the containment hydrogen monitoring system was being effectively controlled through the performance of appropriate preventive maintenance such that the system remained capable of performing its intended function. The inspectors identified that both channels of the containment hydrogen/oxygen (H_2/O_2) analyzers had been out of service since September 7, 2006, due to compressor seal leakage. The inspectors determined that the H_2/O_2 analyzers are within the scope of Entergy's Maintenance Rule program since they are used in the emergency operating procedures. The inspectors noted that, based on the significant unavailability time of both trains, the system should have been in 10 CFR 50.65(a)(1) status with an action plan to improve system performance back to an (a)(2) status. Entergy entered this issue into their corrective action program and changed the priority of the work orders to perform repairs on the H_2/O_2 analyzers.

This inspectors determined that this finding affected the Barrier Integrity cornerstone and was more than minor since it was similar to Example 7.b in IMC 0612, Appendix E, "Examples of Minor Issues." Specifically, Entergy failed to demonstrate effective control of the performance of the H_2/O_2 analyzers and did not place the system in (a)(1) status. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The finding required further evaluation through IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," because it resulted in an actual reduction in the defense-in-depth for the hydrogen control function of the reactor containment. The inspectors determined that this finding was of very low safety significance because it did not affect core damage frequency and the H2/O2 analyzers are not important to large early release frequency.

The inspectors determined this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that equipment and resources were available to assure reliable operation of the H2/O2 analyzers. Specifically, Entergy did not minimize long-standing equipment issues and maintenance deferrals associated with the containment hydrogen monitoring system. (Section 4OA2)

Inspection Report# : 2007002 *(pdf)*

Significance: SL-IV Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE CONTAINMENT CLOSURE EQUIPMENT
The inspectors identified a Severity Level IV NCV of 10 CFR 50.59, "Changes, Tests and Experiments," for failure to obtain a license amendment pursuant to 10 CFR 50.90 prior implementing a change to alter the requirements of a shutdown fission product barrier. The inspectors reviewed Safety Evaluation 04-0732-MD-00-RE R1, "Installation of a Temporary Roll-up Door on the Containment Equipment Hatch," to determine if the conclusion that a licensee amendment was not required was correct. Entergy concluded that the roll-up door was equivalent to the closure plate and, therefore, adequate to close containment as required by the action statement. The inspectors found that the door was not designed to be air-tight; therefore, any radioactive release inside containment would bypass the roll-up door. The inspectors concluded that the roll-up door did not meet the design or licensing basis of the closure plate as described in the Updated Final Safety Analysis Report (UFSAR) and previously approved license amendments. Consequently, Entergy incorrectly concluded that a license amendment pursuant to 50.90 was not required prior to implementing the change. Entergy entered the issue into their corrective action program to evaluate and correct.

The inspectors determined that Entergy changed the requirements for the shutdown fission product barrier (containment) prior to receiving NRC approval. As a result, traditional enforcement was used to evaluate the issue because the deficiency affected the NRC's ability to perform its regulatory function. The severity level of the violation was determined to be Severity Level IV in accordance with example D.5 of Supplement 1 of the NRC Enforcement Policy. Additionally, the issue was determined to be of very low safety significance (Green) based on the low decay heat levels at the time the roll-up door was credited in accordance with the significance determination process described in Inspection Manual Chapter (IMC) 0609 Appendix H, "Containment Integrity."
Inspection Report# : 2006005 *(pdf)*

Emergency Preparedness

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: FIN Finding
INADEQUATE CORRECTIVE ACTIONS FOR FAILURE TO APPROPRIATELY MONITOR SERVICE WATER INTAKE BAY LEVEL
The inspectors identified a Green finding because Entergy failed to take adequate corrective actions for an issue associated with monitoring of service water intake bay level. This deficiency could have prevented identification of entry conditions for an emergency action level. Entergy entered this issue into the corrective action program as CR IP3-2007-00453, and initiated several corrective actions, including plans for enhanced monitoring of service water bay levels, backwashing of trash racks, procedural upgrades, correction of service water bay level instrumentation modification installation, development of modifications for enhanced service water level monitoring equipment, and enhanced inspection and cleaning of intake structure trash racks.

The inspectors determined that this finding was more than minor because it was associated with the Emergency Preparedness cornerstone attribute of facilities and equipment; and, it affected the cornerstone objective of ensuring that a licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, inadequate monitoring of service water intake bay level could have resulted in failure to declare a notification of unusual event (UE). The inspectors reviewed the EAL entry criteria and determined that this performance deficiency did not affect Entergy's ability to declare any event higher than a UE. The inspectors evaluated this finding using IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process," Sheet 1, "Failure to Comply," and determined that it was of very low safety significance because the declaration of a UE based on low service water bay level could have been missed or delayed, consistent with the example provided in the appendix.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not implement effective corrective actions for a previously identified issue associated with inadequate monitoring of service water intake bay level. (Section 1R17)

Inspection Report# : 2007002 *(pdf)*

Occupational Radiation Safety

Significance: G Dec 31, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
FAILURE TO SURVEY AND PROVIDE ACCESS TO AN UNPOSTED HIGH RADIATION AREA
A Green, self-revealing NCV of 10 CFR 20.1501 with respect to 10 CFR 20.1902(b) was identified, in that, Entergy failed to survey radiological condition changes after a plant manipulation that was likely to cause a change in radiological conditions, and this led to the failure to post a plant area as a high radiation area. As a result, two workers were allowed access to an unsurveyed and unposted high radiation area.

The finding is more than minor because it is associated with the Occupational Radiation Safety cornerstone attribute of exposure control and affected the cornerstone objective, because not establishing radiological conditions and commensurate controls after changing plant radiological conditions prior to allowing access to the affected areas can cause increased personnel exposure. The inspectors evaluated this finding using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," and determined that it was of very low safety significance (Green) because it did not involve ALARA planning and controls, an overexposure, a substantial potential for overexposure, or an impaired ability to assess dose. This issue was entered into Entergy's corrective action program and training was provided to the radiation protection staff.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not use a conservative assumption in the decision-making process, in that, the watch radiation protection technician did not question the radiological conditions of the pipe chase area after a change of plant conditions had occurred and did not require a survey of the pipe chase area before authorizing access to personnel.
Inspection Report# : 2006005 *(pdf)*

Significance: G Dec 31, 2006
Identified By: Self-Revealing
Item Type: FIN Finding
UNIT 2 CONTAINMENT SUMP STRAINER MODIFICATION COLLECTIVE EXPOSURE OVERRUNS DUE TO INADEQUATE MOD PREPARATION
A self-revealing finding was identified that involved inadequate modification planning and construction preparations relative to a Unit 2 containment sump strainer modification that resulted in significant unplanned collective exposure (93.7 person-rem compared to a work activity estimate of 10.9 person-rem). Specifically, the actual job site conditions for installation of the containment sump modification were not adequately evaluated with respect to the radiological impact of increased occupancy in high dose rate work areas. This unplanned additional in-field high radiation work resulted in significant unintended exposure that could have been avoided. This issue was entered into Entergy's corrective action program so that lessons learned could be incorporated into the Unit 3 containment sump modification.

The inspectors determined that this finding was more than minor because it was similar to examples 6.a and 6.b of IMC 0612, Appendix E, "Examples of Minor Issues," in that, the issue involved actual collective exposure greater than 5 person-rem and was greater than 50 percent above the estimated or intended exposure; and the majority of the dose overrun was due to activities within Entergy's control. The inspectors evaluated this finding using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," and determined that the finding was of very low safety significance (Green) because it involved an ALARA planning issue, and the 3-year rolling average collective dose for Unit 2 was less than 135 person-rem (73 person-rem average annual exposure for 2003 through 2005).

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not adequately incorporate job site conditions in the work control planning process.
Inspection Report# : 2006005 *(pdf)*

Public Radiation Safety

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Significance: G Dec 05, 2006
Identified By: NRC
Item Type: FIN Finding
FAILURE TO ENTER SAFETY CULTURE ASSESSMENT RESULTS INTO CORRECTIVE ACTION PROGRAM
The NRC inspectors identified a finding when Entergy failed to initiate condition reports in accordance with EN-LI-102, "Corrective Action Process," for the adverse conditions identified in the 2006 Safety Culture Assessment. Consequently, the adverse conditions were not evaluated and appropriate corrective actions were not identified in a timely manner. The contractor who performed the independent safety culture assessment presented the site specific results to Entergy management in June 2006. The negative responses and declining trends identified in the assessment constituted adverse conditions that should have been entered into the corrective action program. At the time of the inspection, Entergy had not initiated condition reports for the assessment results. Consequently, the results had not been fully evaluated to understand the causes and identify appropriate actions to address the identified issues. Additionally, organizations identified by the contractor as needing management attention had not developed departmental action plans at the time of the inspection. Entergy entered this issue into the corrective action program and initiated a learning organization condition report to track development and implementation of action plans to address the assessment results.

The inspectors determined that the finding was more than minor because if left uncorrected it would become a more significant safety concern. Without appropriate action, the weaknesses in the safety culture onsite would continue, increasing the potential that safety issues would not receive the attention warranted by their significance. The finding was not suitable for SDP evaluation, but has been reviewed by NRC management and has been determined to be a finding of very low safety significance. The finding was not greater than very low safety significance because the inspectors did not identify any issues that were not raised which had an actual impact on plant safety or were of more than minor safety significance.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not identify issues with the potential to impact nuclear safety in the corrective action process for evaluation and resolution in a timely manner.
Inspection Report# : 2006006 *(pdf)*

Last modified : August 24, 2007

EXHIBIT LL

Doosan Heavy Industries & Construction Co., Ltd.

www.doosanheavy.com

Presented at the
Burns & Roe 17th Annual Seminar
Powering the Future
March 21, 2007

On-going Projects (I)

Entergy Replacement Reactor Vessel Head

- Customer : Entergy
- Primary Contractor : Westinghouse
- Projects : ANO #2 (Site Delivery: January, 2008)
  Waterford #3 (Site Delivery: February, 2008)
  Indian Point #2 (Site Delivery: October, 2011)
  Indian Point #3 (Site Delivery: October, 2012)
- Scope : Four (4) RRVHs
  Two (2) sets of CRDM (for Indian Point #2 & 3 only)
- Manufacturer : DOOSAN (EMD supplies CRDM as the subsupplier)

Qinshan Phase II #3 Reactor Vessel

- Customer : NPQJVC (Nuclear Power Qinshan Joint Venture Co.)
- Contractors : DOOSAN (#3), CFHI (#4)
- DOOSAN's Scope : One(1) Reactor Vessel & Technical Assistance
- Expected shipping : June, 2008

March 27, 2007

Re: NRC Proposed Rule: Power Reactor Security Requirements (RIN 3150-AG63)

Annette Vietti-Cook, Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attn: Rulemakings and Adjudications Staff
Submitted via e-mail to SECY@nrc.gov

COUNCIL ON INTELLIGENT ENERGY & CONSERVATION POLICY (CIECP) COMMENTS TO PROPOSED RULE 10 CFR PARTS 50, 72 AND 73 REGARDING POWER REACTOR SECURITY REQUIREMENTS AT LICENSED NUCLEAR FACILITIES

Nearly six years after September 11, 2001, the 103 civilian nuclear reactors in the United States are still not in a position to repel attacks by adversaries with capabilities commensurate with those of either the 9/11 terrorists or with enemies of the United States currently operative on the world stage. The present Power Reactor Security Requirements (PRSR) thus fall far short of the actual threat level faced by the U.S. today, much less the escalated level the nation will face as nations such as Russia, China and Iran improve and export nuclear engineering expertise. Indeed, as numerous security experts have pointed out, a terrorist group with access to sympathetic nuclear scientists and engineers would have sufficient sophistication to target the critical systems and weak links of nuclear reactors. The assistance that Pakistani nuclear scientists reportedly offered to Al Qaeda illustrates this threat.

Recent National Intelligence Estimates and National Intelligence Council Reports describe the terrorist threat to the U.S. as real and as having no sign of abatement for many years to come. These reports further warn of a new class of "professionalized" terrorists –in part created by the Iraq war- who must be expected to have strong technical skills and English language proficiency. Such individuals should, in the future, be expected to become major players in international terrorism.

Al Qaeda and other terrorist groups have shown extraordinary tactical ingenuity and a complete lack of reverence for human life. Further there is ample evidence that U.S. nuclear power plants, particularly those sited near metropolitan areas, are viewed as attractive terrorist targets. Notably, the 9/11 Commission learned that the original plan for a terrorist spectacular was for a larger strike, using more planes, and including an attack on nuclear power plants. In an Al-Jazeera broadcast in 2002, one of the planners of 9/11 said that a nuclear plant was the initial target considered. We also know from the 9/11 Commission's investigation that, even after the plot was scaled down, when Mohammed Atta was conducting his surveillance flights he spotted a nuclear power plant (unidentified by name, but obviously the Indian Point nuclear power plant) and came close to redirecting the strike. National Research Council analyses and post-9/11 intelligence has also indicated that the U.S. nuclear infrastructure is viewed as an alluring target for a future terrorist spectacular. As the Chairman of the National Intelligence Council stated in 2004, nuclear power plants "are high on Al Qaeda's targeting list," adding that the methods of Al Qaeda and other terrorist group may be "evolving."

There is, thus, every reason to believe that a sizable, well-planned and orchestrated military operation against a U.S. nuclear facility is well within both present and near-future terrorist intent and capability. In view of these realities, the current proposed PRSR is utterly inadequate.

Consequently, the COUNCIL ON INTELLIGENT ENERGY & CONSERVATION POLICY (CIECP) urges the NRC to address the following realities in its PRSR:

ACTIVE INSIDERS

The voluminous number of security breaches which have occurred at critical infrastructure, including nuclear weapons and power facilities after 9/11 (such as the 16 foreign-born construction workers who were able to gain access to the Y-12 nuclear weapons plant with falsified documentation) demonstrates that nuclear "insiders" must be deemed potential active participants in an attack.

This threat is significantly augmented by nuclear power plant operators' increasing outsourcing of on-site work in order to cut costs.

Contractor oversight failures have been documented by the NRC. For example a December 22, 2003 NRC Special Inspection Report on the Indian Point Nuclear Generating Station in Buchanan, New York (Indian Point) operated by Entergy Nuclear Northeast (Entergy) notes "the common theme of a lack of direct contractor oversight and quality control measures, along with the absence of Entergy subject matter experts to independently assess contracted work activities...." Critically, the risk of sabotage is elevated at all power plants during periods of refueling and major construction work when hundreds of outside contract workers have site access.

The active participation of insiders, including contract workers, in a terrorist offensive need not take place during the time of attack. It may occur days or even many months prior to an attack. In addition to actions such as surveillance of plant schematics, security features and protocols, pre-attack participation may involve the sabotage of critical instrumentation, computers, piping, electronic systems or any number of other components, where such sabotage would likely not be discovered prior to an emergency event.

COMPUTER SYSTEM COMPROMISE

Nuclear power plant computer systems, like those of other critical infrastructure, are subject to a range of vulnerabilities, including power outages, attacks by malicious hackers, viruses and worms. Compromise of integrity may also occur at the level of software development via backdoors written into code or the implantation of logic bombs programmed to shut down a safety system at a particular time.

Many terrorist networks have the resources and technical savvy to wreak havoc. For example, the alleged terrorist, Muhammad Naeem Noor Khan, picked up in Pakistan in 2004, and believed to have links with Al Qaeda, is a computer engineer.

The fact that U.S. nuclear reactors are not impregnable was demonstrated by the penetration of the Slammer worm into the Davis-Besse nuclear facility. That intrusion disabled a safety monitoring system for nearly 5 hours. In addition, computer hackers have broken into U.S. Department of Energy computers. Some of such intrusions were root-level compromises, indicating that hackers had enough access to install viruses.

Computers at nuclear power stations are also vulnerable to acts of sabotage against off-site power transmission, as was evidenced at Indian Point during the 2003 blackout which struck the Northeast. At Indian Point, various computer systems had to be removed from service, including the Critical Function Monitoring System, the Local Area Network, the Safety Assessment System/Emergency Data Display System, the Digital Radiation Monitoring System and the Safety Assessment System.

It is, accordingly, a matter of pressing importance that the NRC engage independent experts to develop a comprehensive computer vulnerability and cyber-attack threat assessment. Such an assessment must evaluate the vulnerability of the full range of nuclear power plant computer systems and the potential consequences of such vulnerabilities. The PRSR must incorporate such findings and include a protocol for quickly detecting such an attack and recovering key computer functions in the event of an attack.

CHEMICAL WEAPONS

The PRSR must fully address the potential consequences of the use of toxic chemicals as part of an attack scenario. There are numerous agents that can be deployed with almost instantaneous effect and can immobilize targets via paralysis, convulsions, blinding, suffocation or death. Such agents could be employed as part of the initialization strategy. For, example, a truck or even large SUV filled with chlorine, boron trifluoride, hydrofluoric acid, liquid ammonia, or any number of other agents could be crashed into a perimeter barrier, with the resulting fumes killing or disabling plant personnel guarding the outdoor area of the facility.

Chemical agents could also be introduced surreptitiously into building ventilation systems. They may also be used strategically to neutralize workers endeavoring to maintain control of the situation.

Many such agents are easy to make and do not require sophisticated delivery systems. Some can be carried in coffee mugs or in vials within body cavities. Phenarsazine chloride, an arsenic derivative, can be transported in minute quantities, even as a powder that can be dusted on paper. It is lethal if burned and even a spoonful can cause immediate extreme irritation of the eyes and breathing passages. A chemical like chloroform ascitone methanol can be transported on filter paper, then combined with a heat source to create an explosion.

CONVENTIONAL WEAPONRY

Intelligence and military analysts have repeatedly warned that extremists in Iraq, the tribal areas of Pakistan and elsewhere are currently developing a high level of military skill and experience. This reality underscores the need for nuclear plants to be able to defend against attackers utilizing the full range of potential weaponry that terrorists are known to be capable of using, including heavy caliber automatic weapons; sniper rifles; shoulder-fired rockets; mortars; platter charges; anti-tank weaponry; bunker busters; shaped charges; rocket-propelled grenades; and high-power explosives.

Numerous weapons systems posing a threat to even the best trained and equipped civilian guard force, as well as to on-site installations, are readily available and easy to transport. To wit:

- Assault rifles and other rapid-fire battlefield weapons such as AK-47's, Uzi's and TEC-9's are freely available in the U.S. A weapon like the SKS 7.62-millimenter semiautomatic assault rifle can be purchased for under $200. In 2005 the Government Accountability Office reported that 47 individuals on a federal terrorism watch list were actually permitted to legally buy guns in 2004.

- A standard M-24 sniper rifle with day and night scope can be carried in a canvas bag and fires 7.62-millimeter ammunition targeting up to 3000 feet

- A .50-caliber Barrett rifle, which can be purchased for $1000 on the internet, weighs a mere 30 lbs and can hit targets up to 6000 feet away with armor-piercing bullets that can blow a hole through a concrete bunker, bring down a helicopter or pierce an armored vehicle.

- A rocket propelled grenade launcher is re-loadable, can fire at the speed of 400 feet per second and can blow a vehicle into the air.

- A TOW missile is an accessible form of military hardware used in over 40 countries and can be fired from a launcher on a flatbed truck. A 1998 test TOW fired into a nuclear waste transport cask (which is more robust than many on-site nuclear waste storage casks) blew out a hole the size of a grapefruit. The Kornet-E missile, developed by the Soviets and sold to Iraq, can travel over 3 miles and cut through over 3 feet of steel. The world's arms market is awash in thousands of Milan missiles. The 60-70 lb Milan missile system has an effective range of over 5000 feet and can blow a hole through more than 3 feet of armor plate.

- The deployment of increasingly powerful and sophisticated explosives, including shaped charges and explosively formed penetrators (or E.F.P.s) by terrorists and insurgents in Iraq show that the explosives use capabilities of enemies of the United States should not be underestimated. Notably, the 18 men arrested in Australia in November 2005, and believed to have been planning an attack on an Australian nuclear reactor, had allegedly been stockpiling materials used to make the explosive triacetone triperoxide, or TATP. Terrorists targeting a U.S. nuclear power plant may very well be able to draw on expertise developed during the Iraq insurgency as well as military experts and rocket scientists from the former Iraq government or from hostile nations such as Iran. In addition, the strategic utility of explosives is magnified when bombers are willing to blow themselves up. Suicide bombers able to gain access to the internal areas of a nuclear power plant during the course of an attack could cause untold destruction.

- Perhaps the most intractable military hardware threat is posed by shoulder-fired missiles such as Stingers, SA-7's, SA-14's and SA-18's. An estimated 500,000 such systems are scattered throughout the world and have been found in the possession of at least 27 terrorist or guerrilla groups. Some can be bought easily on the black market for as little as several thousand dollars each. Critically, shoulder-fired missiles are easy to operate (Al Qaeda training videos offer instruction) and are designed for portability, typically being 5-6 feet long and weighing 35 lbs. They can be transported by and fired from a van, S.U.V., pickup truck or recreational boat. Even a single terrorist armed with a shoulder-fired missile can cause immediate and substantial damage to a targeted structure. Traveling at more than 1,500 miles per hour, a typical shoulder-launched missile has a range of over 12,000 feet. If the target remains intact following the initial strike, the terrorist can attach a new missile tube to the grip stock launcher and fire again.

WATERBORN ATTACKS

Waterborne defenses of nuclear plants adjacent to navigable waterways must be significantly enhanced. Facilities must either be engineered to withstand damage from a waterborne attack or suited with physical barriers that prevent entry to the plant and/or critical cooling intake equipment.

Continual cooling is an essential component of nuclear plant safety. A meltdown can be triggered even at a scrammed reactor if cooling is obstructed. Water intake is also essential to the proper function of spent fuel pools. Yet at certain nuclear plants, cooling systems may be highly vulnerable. At both Indian Point and Millstone Power Station, in particular, water intake pipes have been identified by engineering experts as exposed and susceptible to waterborne sabotage.

One or more boats laden with high energy explosives could severely compromise cooling water intakes easily and quickly. Indian Point, for instance, is located on the banks of the Hudson River in an area heavily trafficked by commercial and recreational vessels. The 900 foot "Exclusion Zone" –marked only by buoys- could be traversed by speed boats in 30 - 40 seconds, well before any Coast Guard or other patrol boat could react. Patrol boats could also be readily taken out by suicide bomber boats crashing into them (in the manner a small explosives laden boat targeted the destroyer the USS Cole in 2000) or by weaponry like shoulder-fired missiles or rocket propelled grenades.

AERIAL ASSAULT

According to a terrorist "threat matrix" issued by the National Research Council and the National Academies of Sciences and Engineering following the September 2001 attack, "Nuclear power plants may present a tempting high-visibility target for terrorist attack, and the potential for a September 11-type surprise attack in the near term using U.S. assets such as airplanes appears to be high."

In March 2005, a joint FBI and Department of Homeland Security assessment stated that commercial airlines are "likely to remain a target and a platform for terrorists" and that "the largely unregulated" area of general aviation (which includes corporate jets, private airplanes, cargo planes, and chartered flights) remains especially vulnerable. The assessment further noted that Al Qaeda has "considered the use of helicopters as an alternative to recruiting operatives for fixed-wing operations," adding that the maneuverability and "non-threatening appearance" of helicopters, even when flying at low altitudes, makes them "attractive targets for use during suicide attacks or as a medium for the spraying of toxins on targets below."

The vulnerability of nuclear power plants to malevolent airborne attack is detailed extensively in the Petition filed by the National Whistleblower Center and Randy Robarge in 2002 pursuant to 10 CFR Sec. 2.206. A number of studies of the issue are also reviewed in Appendix A to these Comments. The particular vulnerability of nuclear spent fuel pools to this kind of attack is detailed in the January 2003 report of Dr. Gordon Thompson, director of the Institute for Resource and Security Studies entitled "Robust Storage of Spent Nuclear Fuel: A Neglected Issue of Homeland Security" and in the findings of a multi-institution team study led by Frank N. Von Hippel, a physicist and co-director of the Program on Science and Global Security at Princeton University and published in the spring 2003 edition of the Princeton journal *Science and Global Security* under the title "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States." It is worthy of note that, even post-9/11, general aviation aircraft have circled or flown closely over commercial nuclear facilities without military interception.

The NRC's sole present strategy for averting a kamikaze attack upon a nuclear power plant is reliance upon aviation security upgrades implemented by the Transportation Security Administration and the Federal Aviation Administration and faith that U.S. intelligence will provide ample warning.

It is this kind of governmental agency pass-the-buck mindset that brought the nation Katrina.

The NRC's conjecture also betrays a reality disconnect reminiscent of the federal response to Katrina. Since 2001 there have been numerous breaches of airport security throughout the nation. Notably, in late 2005, there were three serious security breaches at Newark International Airport, one of the points of departure used by the September 11 hijackers. The most serious occurred on November 12, 2005, when a man driving a large S.U.V. barreled through the armed security checkpoint and drove in a secured area for 45 minutes before being found by NY/NJ Port Authority officers. Just this year, gaping holes in airport security were exposed when workers with access to secure areas were able to carry firearms in their carry-on bags onto a commercial jet departing from Florida.

The PRSR must furthermore be upgraded to include high-speed attack by a jumbo jet of the maximum size anticipated to be in commercial use (such as the expanded version of the Boeing 747 and the Airbus A380) as well as unexpected attack by general aviation aircraft and helicopters. The PRSR must contemplate all such aircraft to be fully loaded, fueled and armed with explosives.

It is essential that the PRSR address not only the direct effect of impact, but the full potential aftereffects of (A) induced vibrations; (B) dislodged debris falling onto sensitive equipment; (C) a fuel fire; and (D) the combustion of aerosolized fuel (especially in combination with pre-existing on-site gases such as hydrogen).

The PRSR must further take into consideration the cascading consequences of aerial assault on the full spectrum of plant installations. Inarguably, there is a wide range of on-site structures, not within hardened containment, that are critical to the safe operation of a nuclear plant. Spent fuel pools are of particular concern because the disposition of water could uncover the fuel. If plant workers are unable to effectuate replacement of the water (either because of fire or because they are otherwise incapacitated), experts warn, an exothermic reaction could cause the zirconium clad spent fuel rods to ignite a nuclear waste conflagration that would very likely spew the entire radioactive contents of the spent fuel pool into the atmosphere.

Without question, hardening a nuclear power plant against aerial threat will necessitate significant upgrades in plant fortification. However even relatively modest measures such as the installation of Beamhenge and the placement of all sufficiently cooled spent fuel into Hardened On-Site Storage Systems (known as H.O.S.S.) would add measurable protection.

STRATEGIC USES OF RIGS, TRUCKS AND S.U.V.'S

In June 1991, the NRC denied the truck bomb petition of the Committee to Bridge the Gap and the Nuclear Information Resource Service, on the grounds that it was not realistic to believe a truck bomb would be employed in the U.S. Two years later, on February 26, 1993, terrorists drove a rented van packed with explosives into the underground garage of the World Trade Center, lighted a fuse and fled. Just a couple of weeks before that, a mentally unstable individual crashed his station wagon through the gates of the protected area of the Three Mile Island nuclear power station and evaded security for several hours before finally wrecking his vehicle by crashing into the turbine building. Thereafter, the NRC reconsidered its earlier assessment and has, on a number of occasions, upgraded reactor security standard to include some protections against land vehicles. Such upgrades, however, are insufficient in a post-9/11 world.

Large Sport Utility Vehicles and pickup trucks on the road today can weigh over 8 tons, loaded, and -as do commercial vans- have considerably carrying capacity. Such vehicles could be used strategically in a number of ways.

The first is as a mobile short range projectile bomb. A large, heavy vehicle packed with high explosives, even if not successful in penetrating concrete barriers, could result in the death or incapacitation of large numbers of plant workers, including security, personnel. Such casualties would be particularly likely to materialize if the vehicle bomb followed a previous diversionary event intended to draw security personnel to the plant perimeter.

The second is as a transport vehicle for one team of attackers who are themselves armed or who wear explosive belts and could then themselves penetrate other areas of the facility. A terrorist wearing an explosive body belt can, in effect, be a precision guided weapon.

The third and fourth scenarios are variations of the first two, with chemical agents substituted for or combined with explosives. (Indeed, insurgents in Iraq are increasingly combining explosives with chlorine gas and other chemical payloads in truck bomb detonations.) One or two such vehicles packed with the right toxins, could be expected to kill or disable a substantial number of workers, again, especially if the release followed a prior event which drew security personnel to the area, or simply to areas outside facility enclosures. Certain toxins can be lethal to anyone within miles. Using such agents, attackers wearing protective gear could then gain access to other areas of the facility.

A fifth tactical use of vehicles would not even occur on site. Vehicles carrying explosives and/or chemical agents could be set off at critical regional transportation arteries such as major bridges, tunnels and highways. Notably, such incidents could be staged in a way that would not even alert authorities to the onset of terrorist activity. In the New York metropolitan region in which Indian Point is sited, for example, a series of major accidents occurring at or about the same time would not be an unusual occurrence. In fact, on July 25, 2003, the very day the Federal Emergency Management Agency declared that the Indian Point emergency plan provided "adequate" assurance of protection to the public, the entire New York metropolitan region was brought to a virtual traffic standstill after a tractor-trailer hit a beam on the George Washington Bridge and burst into flames, several minor accidents and a car fire took place on Interstate 95, and a truck got jammed under an overpass of the Hutchinson River Parkway. In 2006, a tanker truck carrying 8000 gallons of gasoline overturned on one of New York City's busiest highways, igniting a blaze that burned for hours and weakening the steel beams of an above bridge. Earlier this month a liquid propane explosion closed a 23 mile stretch of the New York State Thruway for hours, while firefighters had to stand by and watch the fire burn out because it was too hot to approach.

The staging of a couple of incidents like those just noted, combined with an "accident" involving a tanker carrying hazardous gasses or liquids like liquefied ammonia, propane, chlorine, or vinyl chloride, prior to an assault would almost assuredly forestall the provision of outside assistance to a nuclear facility under attack.

PLANTS MUST BE ABLE TO MOUNT A FULL DEFENSE WITHOUT RELIANCE ON OUTSIDE ASSISTANCE

Whether or not an attack employs strategies designed to obstruct regional transportation routes, numerous studies and the actual events of 9/11, Katrina, and Rita (as well as relatively minor events such as the January 18, 2006 wind storm in NY) demonstrate beyond cavil that first responder forces and the National Guard do not have the resources, manpower, equipment or communications capabilities to swiftly and adequately respond to a major assault on a nuclear facility. Just this very month, a report of the Commission on the National Guard and Reserves detailed the ongoing problem of inadequate human, equipment, communications and financial resources plaguing the National Guard. This report calls into question the ability of the government to bring all necessary assets to bear in the immediate aftermath of a major domestic incident.

In some regions - most notably the New York Metropolitan region, in which Indian Point is sited – roadway logistics and regular congestion alone would likely prevent assisting forces from reaching a nuclear plant under attack in time. It bears mention that SWAT team assembly takes approximately 2 hours, whereas an assault could be over in a matter of minutes.

It is accordingly crucial that the NRC cedes the faulty assumption that plant personnel need only fend off attackers until law enforcement or military aid arrives. The fact that most regional first responders have little detailed knowledge of either the operational or internal layout of nuclear facilities further testifies to the folly of reliance upon the "cavalry".

ELEVATED VULNERABILITY TO INFILTRATION DURING EVENT

During a crisis event at a nuclear plant there also exists an elevated threat of infiltration by terrorists posing as first responders or National Guard. And in fact the imposter tactic has been used by terrorists in recent years with substantial success.

Terrorists disguised as firefighters could take particularly strong advantage of this stratagem. Outside firefighters often respond to fires at nuclear power plants and many attack scenarios would be expected to involve fire. Firefighters would presumptively be seen as benign by plant personnel and would have a legitimate reason to move throughout a facility and "check" components such as electrical wiring. Moreover, bulky firefighter uniforms and equipment can hold and hide a host of articles that could be used for destructive purposes.

DEFENSE AGAINST A SIZABLE MULTI-TEAM, MULTI-DIRECTIONAL FORCE

In January 1991, the Nuclear Information Resource Service and the Committee to Bridge the Gap filed a joint Petition with the NRC requesting, *inter alia,* that the DBT be upgraded to 20 external attackers. The NRC rejected the petition in June 1991, asserting that an attack involving more than 3 assailants was unrealistic.

September 11 was a demonstration of the profound limitations of governmental foresight.

The September 11 plot involved 20 attackers (although only 19 were ultimately able to participate). The tragic 2004 siege at a school in Belsan, Russia involved more than 30 armed terrorists. It should be beyond question at this point that a terrorist attack could involve scores of attackers.

Accordingly, the PRSR must assume at least two dozen attackers. Lessons learned from 9/11 and the many multiple coordinated terrorist actions that have transpired in Europe, Asia and the Middle East since then, also mandate the premise that attackers will act in several teams and that some of those teams may be sizable.

Any carefully planned attack on a nuclear facility by knowledgeable individuals, would also involve several different *modus operandi.* The PRSR should therefore take into account the consequences of near-simultaneous damage to different plant installations, systems and personnel (e.g., the effect of a small explosive-laden plane diving into the roof of a spent fuel pool coupled with the waterborne sabotage of the spent fuel pool intake system).

A COORDINATED ATTACK ON MULTIPLE ON AND OFF-SITE TARGETS

A related point is that, following 9/11, the NRC can no longer ignore the very real possibility that an attack on a nuclear power plant would occur commensurate with an attack on other regional infrastructure such as chemical plants and bridges. A coordinated attack designed to effectively eradicate a region would very likely preliminarily target communication, electrical power and/or transportation infrastructures. This would ensure that (A) the targeted region is reduced to mass confusion, (B) local and federal officials and responders would be overwhelmed, and (C) law enforcement and other first responders would be impeded from gaining access to the nuclear plant site.

Certain areas of the U.S. offer a plethora of target opportunities and thus are particularly vulnerable to multiple target scenarios. Prime among them is the greater New York Metropolitan area (already in the terrorists' crosshairs) which contains numerous national landmarks, corporate headquarters, reservoirs, bridges, airports, transportation arteries and hazardous chemical plants, all in near vicinity to Indian Point, a mere 24 miles north of New York City.

A CREDIBLE NUCLEAR PLANT SECURITY FORCE TESTING PROGRAM

The deficiencies, failures, and chicanery that have long plagued the various manifestations of nuclear power industry security drills and force-on-force (FOF) testing have been exhaustively documented in recent years. Noteworthy investigations in this regard have been conducted by the Project on Government Oversight (augmented by testimony provided in 2002 Senate Environment and Public Works Committee hearings) and the United States General Accounting Office (which reported its findings in a September 2003 report entitled "Oversight of Security at Commercial Nuclear Power Plants Needs to Be Strengthened") as well as by the press. Problems with the FOF program are also addressed in the July 2004 Petition for Rulemaking to amend 10 CFR Part 73 to upgrade the DBT filed by the Committee to Bridge the Gap and the Comments on the DBT filed in 2006 by the Union of Concerned Scientists. CIECP fully endorses the recommendations made in previous filings by the Committee to Bridge the Gap and the Union of Concerned Scientists.

CIECP urges the NRC in the strongest possible terms to upgrade drills and testing protocols to remedy the flaws that are a matter of public record and to take into account the realities noted herein. FOF tests must be sufficiently challenging to provide high confidence in the defensive capabilities of the security forces at the nation's 103 nuclear power plants. One clear failing of the FOF program to date has been the giving of excessive warning regarding upcoming tests. While some notice is necessary, one week should suffice. In addition, staff assignments should be frozen on the day of notice. This would eliminate the all too common practice of substituting a plant's most fit and accomplished security personnel in place of underachievers.

It is also critical that drills and the FOF program be revamped to eliminate manifest conflicts of interest. Examples of blatant conflicts of interest include: (1) The NRC allowing the nuclear industry's lobbying arm, the Nuclear Energy Institute (NEI) to award a FOF contract; and (2) The NEI, with NRC approval, then selecting Wackenhut, a corporation which contracts security guards to nuclear power plants in the U.S., to also be the contractor that supplies the mock adversary teams for the FOF tests.

Such problems have reduced the value of testing to the point where the FOF program lacks public confidence. The program must be redesigned and monitored by an independent entity such as the very capable U.S. military.

HIGH TARGET APPEAL REACTORS

Prior terrorist attacks and plots against the U.S. have focused on major cities. It is a matter of fundamental logic that plants sited in highly populated metropolitan areas, particularly those with high symbolic value, face the greatest risk of being selected as a target.

It is thus imperative that the PRSR be modified to mandate a customized approach to high target nuclear facilities.

SITE-SPECIFIC SAFETY-RELATED VULNERABILITIES

It is highly unrealistic to exclude from the PRSR calculus the reality of aging structures, deteriorated conditions and compromised systems that exist at various nuclear power plants in the U.S. A facility-customized approach must be taken which adds problems which are known or reasonably suspected and which could have a significant effect upon the ability of plant operators to maintain control during a major incident into the security equation.

Prime among factors which may be site-specific are:

- Corrosion and Embrittlement: For example, a risk of corrosion of the steel liner of the reactor containment at the Oyster Creek Nuclear Generating Station (Oyster Creek) was recently identified. A qualified corrosion expert has warned that the risk may be high enough to cause buckling and collapse. Manifestly, corrosion or embrittlement-weakened structures and components are more vulnerable to the effects of heat and combustion.

- Vulnerability to Fire: Fire detection and suppression equipment and fire barriers are crucial to reactor safety. Over 20 years ago a worker at the Brown's Ferry Unit 1 reactor accidentally started a fire which destroyed emergency cooling systems and severely compromised the plant's ability to monitor its condition. In response, the NRC increased fire safety standards. In recent years, the NRC has effectively relaxed those standards. This is exceedingly unwise. During the chaos and threat level that would surely exist during a terrorist attack, human beings cannot be presumed to be able to take the actions necessary to protect critical systems from fire. The systems themselves must have integral safeguards. Yet plants such as Arkansas Nuclear One, Catawba, Ginna, H.B.

Robinson, Indian Point, James A. Fitzpatrick, McGuire, Shearon Harris, Vermont Yankee and Waterford have been identified as having fire barrier wrap systems that failed fire tests. Fireproofing problems such as these jeopardize safe shutdown and must be recognized as a degradation of defense-in-depth protection. In addition, any plant fire hazard analyses must assume damage to multiple rooms and multiple structures, a circumstance that could easily result from an aircraft impact.

- Integrity of Structures that Support Mobility: While the focus of NRC regulatory review is on structures and equipment directly related to safe operational function, the conditions that may prevail during an assault would likely require plant personnel to be able to move rapidly throughout the facility. The evaluation of the reliability of structural features such as stairways (which might buckle or melt during a fire) is accordingly critical.

- Electrical System Problems: In 2003, a cable failure knocked out power to approximately half the safety systems at Oyster Creek, including security cameras, alarms, sensors, pumps and valves. In February 2003, all 4 of the backup generators at Fermi became simultaneously inoperable. In December 2001, Indian Point reactor 2 lost power due to a malfunction of the turbine, then lost back-up power to the reactor coolant system because of a second electrical failure. During the August 2003 blackout that struck the Northeast, following the loss of off-site power, two of Indian Point's emergency backup generators (both of which had been previously flagged as having problems) failed to operate. In view of the severe consequences failures such as these could have were they to occur during a major incident, known plant electrical system vulnerabilities must be taken into consideration.

- Cooling System Problems: Cooling system problems and design deficiencies have plagued a number of plants in recent years. In some cases the NRC has allowed plants to operate for long periods with compromised emergency cooling systems. For example, the Salem nuclear power station had experienced two years of repeated malfunctions of its high-pressure coolant-injection system prior to the time, in October 2003, when operators unsuccessfully tried to use it to stabilize water levels following a steam pipe burst. And the NRC has allowed reactors with emergency sump pumps flagged as likely to become clogged and inoperative to remain in operation for many years without repair. The Los Alamos National Laboratory, for instance, concluded that the sump pumps at Indian Point reactors 2 and 3 could become clogged in as little as 23 minutes and 14 minutes, respectively. While, upgrades are being made, the failure of the NRC to mandate immediate correction of cooling system vulnerabilities calls its oversight capabilities seriously into question. Indeed the functional declination of critical systems must be deemed a constituent element of site-specific PRSR analyses.

ELIMINATE COMMERCIAL CONSIDERATIONS FROM THE PRSR CALCULUS

The commercial interests of the nuclear industry are of valid concern to nuclear utilities and the NEI; they should not be of concern to the NRC. There is no justification for jeopardizing national security and the health and safety of the public - even to the smallest degree – to safeguard corporate profits.

The NRC has stated that its promulgated security standards are based upon the analysis of the largest threat against which a "**private security force could reasonably be expected to defend**" [*emphasis added*] 70 FR 67385.

Both the NRC and the industry have acknowledged that, in their estimation, a private guard force should not be reasonably expected to defend against a 9/11-type attack involving aircraft. Such an attack, apparently, is deemed to fall under the loophole of 10 CFR Sec. 50.13, which exempts reactor operators from defending against "an enemy of the United States, a foreign government or other person". The perimeter of this "enemy of the United States provision has never been defined, so there is no way to know how far it extends. However, it is abundantly clear from the public record that the NRC has drawn the line at point where the profit margins of nuclear power operators might be significantly affected. Unfortunately, the terrorists are constrained by no such boundary.

Congress has charged the NRC with the obligation to protect the public health and safety. This must not be viewed simply as a guideline; it must be viewed as an uncompromised mandate.

If the NRC does not believe its licensees can afford the security upgrades necessary to protect the nation's nuclear reactors against the full potential threat, it must act with forthrightness and publicly demand that the Department of Homeland Security or the U.S. military assume responsibility for domestic nuclear power plant security.

CONCLUSION

The 9/11 Commission observed: "Across the government, there were failures of imagination, policy, capabilities...The most important failure was one of imagination. We do not believe leaders understood the gravity of the threat."

As a public interest group we ask: What needs to happen before the gravity of the threat is not only understood, but acted upon?

Respectfully submitted,

COUNCIL ON INTELLIGENT ENERGY
& CONSERVATION POLICY
(New York)
By

Michel C. Lee, Esq.
Chairman

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<u>APPENDIX A</u>

Since September 11, 2001, there has been much speculation about the vulnerability of nuclear power plants to aerial attack. Certainty, however, is in short supply.

What is known is that none of the nuclear reactors presently operational in the United States were built to withstand the crash of a jumbo jet, much less the crash of super jumbo such as the A380 which will take to the air weighing 1.2 million pounds, has a wingspan almost as long as a football field, is 8 stories tall, and is 3 times as large as the 767s that brought down the Twin Towers.

Nevertheless studies that have addressed the prospect of planes hitting nuclear plants include the following:

<u>1974</u>: To date the only published peer reviewed study on the vulnerability of U.S. nuclear power plants was conducted by General Electric, the leading builder of nuclear plants, and published in the industry journal *Nuclear Safety.* GE looked at accidents – not terror attacks – and concluded that were a "heavy" airliner to hit a reactor building in the right place, it would almost certainly rip it apart. Such a hit would also most likely damage the reactor core and both the cooling and emergency cooling systems. [NOTE: The GE study defined a "heavy" plane as one weighing more than 6 tons. The Boeing 757 which gouged a 100 foot gash through the reinforced concrete of the Pentagon weighed between 80 and 100 tons. A fully loaded 767 weighs over 200 tons. The Airbus 380, expected to be launched into commercial use later this year, takes to the air weighing 1.2 million pounds, hundreds of thousands of pounds heavier than the Boeing 747, the current jumbo of the sky.]

<u>1982</u>: A technical report (previously publicly available) of a study conducted by the U.S. Army Corps of Engineers at the NRC's behest focused on plane crash analyses at the Argonne National Laboratory. The Corps concluded that planes traveling at a speed of over 466 mph would crash through the average reactor containment structure noting "account has been taken of the internal concrete wall which acts as a missile barrier...It would appear, however, that this is too optimistic since vaporized fuel, hot gaseous reaction products, and to a certain extent portions of liquid fuel streams will flow around such obstructions and overwhelm internal defenses...." [NOTE: An FBI analysis estimated that American Airlines Flight 11, which hit the north tower of the World Trade Center, was traveling at a speed of 494 mph, and that United Airlines Flight 175, which hit the south tower, was traveling at 586 mph, a speed far exceeding its design limit for the altitude.]

<u>2000</u>: A NRC study published less than a year before September 11 calculated that 1 out 2 commercial airplanes flying in the year 2000 were large enough to penetrate even a 5 foot thick reinforced concrete wall 45% of the time. Specifically, the study states, "aircraft damage can affect the structural integrity of the spent fuel pool or the availability of nearby support systems, such as power supplies, heat exchangers, or water makeup sources and may also affect recovery actions...It is estimated that half the commercial

aircraft now flying are large enough to penetrate the 5 foot thick reinforced concrete walls." [NOTE: The thickness of the top of certain reactor domes is 3 and-a-half feet.]

<u>2002</u>: The German Reactor Safety Organization (GRS) a scientific-technical research group that works primarily for nuclear regulators in Germany conducted an extremely detailed study that determined that terrorists can, with a strategically targeted airplane crash, initiate a nuclear accident. (A secret Ministry document that summarized the report was leaked to the German and Austrian press and subsequently translated into English.) The GRS study used dynamic computation modeling that looked at the potential consequences of a wide range of impact possibilities on different plant equipment and installations. Different types of airplanes, velocities, angles of impact, weight loads and fuel effects were considered, as were various sequences of events. Aside from the basic finding of vulnerability, the GRS study is significant for recognizing the limitations of even its highly complex analyses. Key unknowns include the impacts of fire loads on many kind of materials and equipment as well as the behaviors of various combustive materials under the conditions of a plane crash.

<u>2004</u>: In 2004 the U.K. Parliamentary Office of Science and Technology (OST) issued a secret report on the risks of terrorist attacks on nuclear facilities to the U.K. House of Commons Defense Committee. The OST report was leaked to the magazine *New Scientist*, which reported the OST conclusion that a large plane crash into a nuclear reactor could release as much radiation as the1986 accident at Chernobyl, while a crash into the nuclear waste tanks at the U.K.'s Sellafield facility could cause several million fatalities.

From these studies it is clear that there exists a reasonable basis for concern regarding malevolent deployment of aircraft against nuclear power facilities.

It should also be evident that all studies on this topic are, in substance, educated conjecture. The current state of computer modeling is not up to analyzing the full range of physical and chemical interactions that could occur under the incalculable range of different kinds of aircraft, approaching at different angles, at different speeds, hitting different structures, which all have facility-unique room and equipment layouts, and different substance, chemical, and ventilation-related conditions.

A lesson in the unpredictable consequences of airplane crashes was brought home on September 11 (when even the 47 story tall 7 World Trade Center that was not struck collapsed for reasons engineers have yet to fully determine). A lesson in the limitations of advanced computer modeling can also be learned from the Columbia space shuttle disaster.

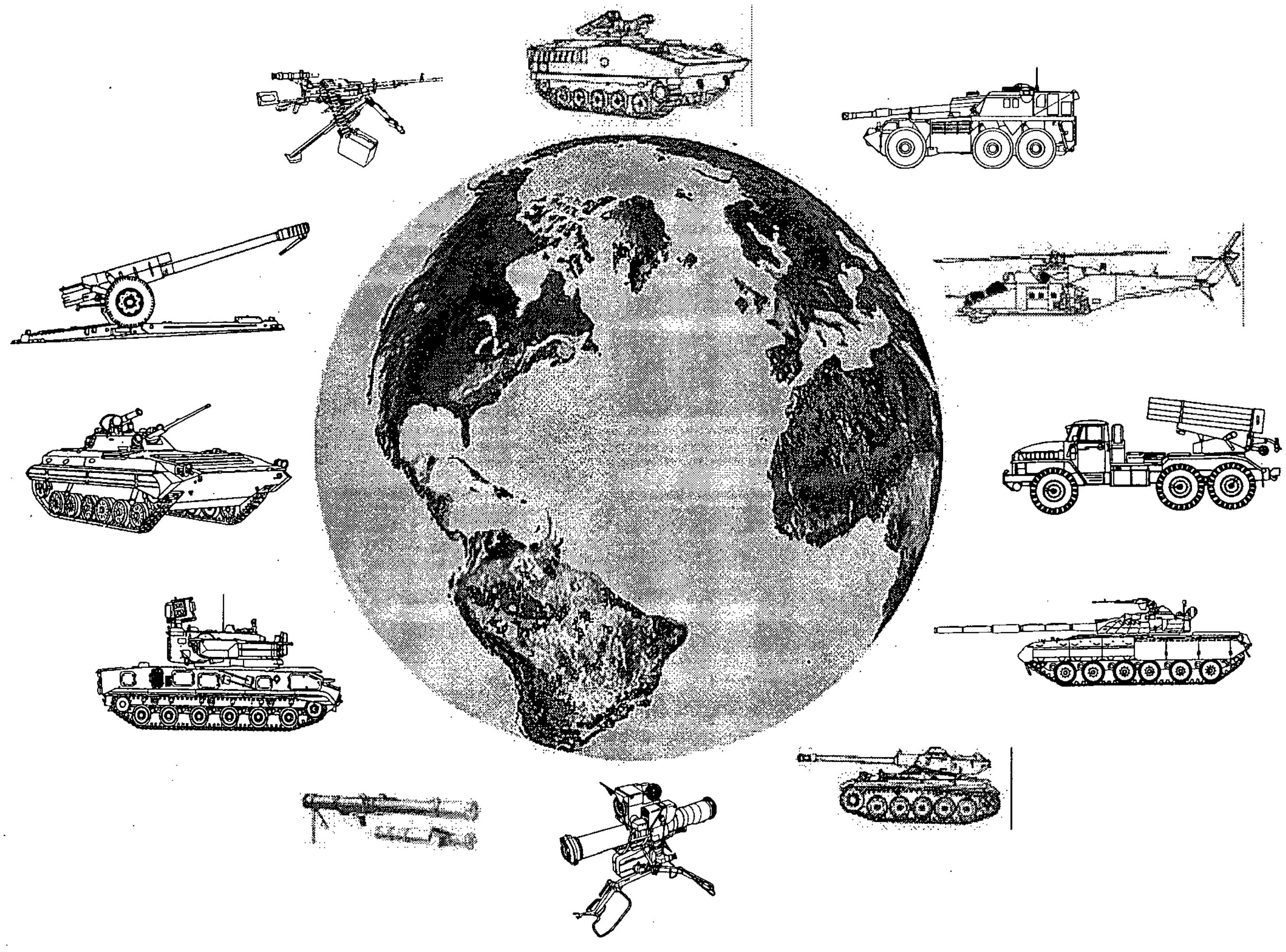
[~DBT and PRSR]

EXHIBIT FF

EXHIBIT NN

EXHIBIT OO

WORLDWIDE EQUIPMENT GUIDE



TRADOC DCSINT Threat Support Directorate

DISTRIBUTION RESTRICTION: Approved for public release; distribution unlimited.

Introduction

This Worldwide Equipment Guide (WEG) serves as an interim guide for use in training, simulations, and modeling until the publication of FM 100-65, *Capabilities-Based Opposing Force: Worldwide Equipment Guide.* The WEG is designed for use with the FM 100-60 series of capabilities-based opposing force field manuals. It provides the basic characteristics of selected equipment and weapons systems readily available to the capabilities-based OPFOR, and generally listed in either *FM 100-61, Armor- and Mechanized-Based Opposing Force: Organization Guide* or *FM 100-63, Infantry-Based Opposing Force: Organization Guide*. Selected weapons systems and equipment are included in the categories of infantry weapons, infantry vehicles, reconnaissance vehicles, tanks/assault vehicles, antitank, artillery, air defense, engineer and logistic systems, and rotary-wing aircraft.

The pages in this WEG are designed for insertion into loose-leaf notebooks. Since this guide does not include all possible OPFOR systems identified in the OPFOR field manuals, equipment sheets covering additional systems not contained in this initial issue will be published periodically. Systems selected will be keyed directly to the baseline equipment contained in the 100-60 series and substitute systems found in the appropriate substitution matrix. The WEG is scheduled for eventual publication on the worldwide web for use by authorized government organizations.

WORLDWIDE OPFOR EQUIPMENT

Due to the proliferation of weapons through sales and resale, wartime capture, and licensed or unlicensed production of major end items, distinctions between equipment as friendly or OPFOR have blurred. Sales of upgrade equipment and kits for application to weapon systems have further blurred distinctions between old or obsolete systems and modern systems. This WEG describes base models listed in the FMs or upgrades of those base models, which reflect current capabilities. Many less common variants and upgrades are also addressed.

HOW TO USE THIS GUIDE

The WEG is organized by categories of equipment, in chapters. The format of the equipment pages is basically a listing of parametric data. This permits updating on a standardized basis as data becomes available. For meanings of acronyms and terms, see the Glossary. Please note that although most terms are the same as U.S. terminology, some reflect non-U.S. concepts and are not comparable or measurable against U.S. standards. For example, if an OPFOR armor penetration figure does not say RHA (rolled homogeneous armor), do not assume that is the standard for the figure. Please consult the Glossary often. If questions remain, contact this office.

System names refer back to the field manuals. However, they also reflect intelligence community changes in naming methods. Alternative Designations include the manufacturer's name, as well as U.S./NATO designators. Note also that the WEG focuses on the complete weapon system (e.g., AT-4/5 antitank guided missile launcher or 9P148 ATGM launcher vehicle), versus a component or munition (9P135 launcher assembly or AT-4/5 ATGM).

Common and consistent technical notes and parameters are used in chapters 2 through 7, since the systems contained in those chapters have similar weapon and automotive technologies. Chapters 1 (Infantry Weapons), 8 (Engineer and Logistics) and 9 (Rotary-wing Aircraft) offer systems that have many unique parameters and therefore may not be consistent with those in other chapters.

We solicit your assistance in finding unclassified information which can be certified for use. Questions and comments on systems data should be addressed to the authors noted for each chapter. For questions concerning distribution to U.S. government organizations, please contact the local publications clerk, and:

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Units of Measure

The following symbols and abbreviations are used in this guide.

Unit of Measure	Parameter
(°)	degrees of slope/gradient, elevation, traverse
cal	caliber—(tube length in multiples of cannon bore)
GHz	gigahertz—frequency (GHz = 1 billion hertz)
hp	horsepower (kWx1.341 = hp)
Hz	hertz—unit of frequency
kg	kilogram(s) (2.2 lb.)
kg/cm^2	kg per square centimeter—pressure
km	kilometer(s)
km/h	km per hour
kW	kilowatt(s) (1 kW = 1,000 watts)
liters	liters—liquid measurement (1 gal. = 3.785 liters)
m	meter(s)—if over 1 meter use meters; if under use mm
m^3	cubic meter(s)
m^3/hr	cubic meters per hour—earth moving capacity
m/hr	meters per hour—operating speed (earth moving)
MHz	megahertz—frequency (MHz = 1 million hertz)
min	minute(s)
mm	millimeter(s)
m/s	meters per second—velocity
mt	metric ton(s) (mt = 1,000 kg)
rd/min	rounds per minute—rate of fire
RHAe	rolled homogeneous armor (equivalent)
shp	shaft horsepower—helicopter engines (kWx1.341 = shp)
μm	micron/micrometer—wavelength for lasers, etc.

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Threat Support Directorate
Ft. Leavenworth, KS, 21 January 1999

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Chapter 1
Infantry Weapons

This chapter provides the basic characteristics of selected infantry weapons either in use or readily available to the OPFOR and therefore likely to be encountered by U.S. forces in varying levels of conflict. The selection of weapons is not intended to be all inclusive, rather a representative sampling of weapons and equipment supporting various military capabilities.

This chapter is divided into two categories—***small arms*** and ***recoilless weapons***. ***Small arms*** covers, in order, assault rifles, under-barrel grenade launchers, light machineguns, general purpose machineguns, heavy machineguns, and automatic grenade launchers. The second category, ***recoilless weapons***, contains the US 106-mm Recoilless Rifle M40 and the Russian 73-mm Recoilless Gun SPG-9. This category also covers a rapidly growing segment of shoulder-fired (unguided) infantry weapons. While originally limited to shoulder-fired unguided antitank weapons such as the Russian 40-mm Antitank Grenade Launcher RPG-7, the utility of shoulder-fired weapons has expanded to include multi-purpose systems such as the Swedish 84-mm Recoilless Rifle Carl Gustaf M2. This field of weapons is often labeled "antitank" and also includes "bunker-buster" warheads, and weapons fired from close spaces such as the German 67-mm Disposable Antitank Grenade Launcher Armbrust.

Another emerging battle-tested, lethal, shoulder-fired weapon is the Russian Infantry Rocket Flame Weapon RPO-A Series (RPO-A/D/Z) capable of firing either a smoke, incendiary, or a thermobaric warhead to 600 meters. At 200 meters it is accurate to 0.5 m^2. The thermobaric warhead has a demolition effect corresponding to a round of 122-mm HE artillery. Due to the relative low cost, availability, versatility, transportability, trainability, and lethality of this category of infantry weapons, trainers should expect to encounter these systems in larger numbers with increasing levels of lethality, penetration, and utility. For information on guided antitank weapon systems see Chapter 5.

Questions and comments on data listed in this chapter should be addressed to:

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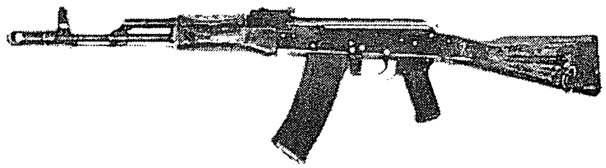
Russian 5.45-mm Assault Rifle AK-74



Ammunition Types	Typical Combat Load
5.45-mm cartridge	300
Ball	
Ball-tracer	
Incendiary-T	
AP	

SYSTEM

Alternative Designations: INA
Date of Introduction: 1974
Proliferation: Widespread

Description:
Weight (kg):
Loaded (with magazine): 3.95
Empty (w/o magazine): 3.4
Length (mm):
Overall: 880 (937 including muzzle brake)
Barrel: 415
Rate of Fire (rd/min):
Cyclic: 600
Practical:
Automatic: 100
Semiautomatic: 40
Operation: Gas
Feed: 30-rd detachable box magazine (40-rd used by RPK-74 LMG is interchangeable)
Fire Mode: Selective, automatic or semi-automatic

SIGHTS

Name: INA
Type: Fore, pillar; rear, U-notch
Magnification: None
Night Sights Available: Yes. AK-74M N3 mounts an NSPU-3

VARIANTS

AKS-74: Folding-stock version with a Y-shaped, tubular stock.
AK-74M: Improves the basic AK-74 design by adding a folding plastic stock, an improved mount for night vision or other sights.
AKS-74U: Submachinegun: modified version with a much shorter barrel (207-mm) and a conical flash suppressor instead of a muzzle break. Its overall length is 492 with stock folded.
AK-101: 5.56x45-mm (NATO) variant of the AK-74M.
AK-102: 5.56x45-mm (NATO) short-barrel (314-mm) variant of the AK-74M.
AK-103: 7.62x39-mm variant of the AK-74M.
AK-104: 7.62x39-mm short-barrel (314-mm) variant of the AK-74M.
AK-105: 5.45x39-mm short-barrel (314-mm) variant of the AK-74M.

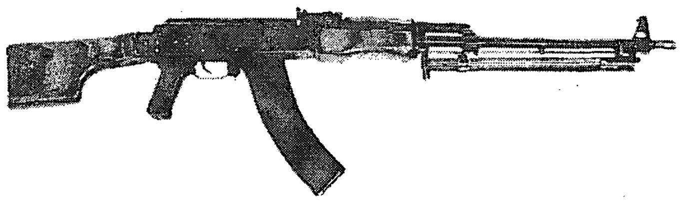
AMMUNITION

Name: 7N6
Caliber/length: 5.45x39-mm
Type: Ball
Range (m):
Effective: 500
Maximum: 800
Armor Penetration: INA
Muzzle Velocity (m/s): 880

Name: 7N10
Caliber/length: 5.45x39-mm
Type: Armor piercing
Range (m):
Effective: INA for AK-74 (800 for RPK-74)
Armor Penetration (mm): 16 @ 100 m 80% of time
Muzzle Velocity (m/s): INA for AK-74 (960 for RPK-74)

NOTES
The AK-74 is basically an AKM rechambered and rebored to fire a 5.45-mm cartridge. The AK-74 can mount a 40-mm under-barrel grenade launcher and a passive image intensifier night sight. The AK-74 is also the basis for other 5.45-mm infantry weapons including the RPK-74 light machinegun.

Russian 5.45-mm Light Machinegun RPK-74



Ammunition Types	Typical Combat Load
5.45-mm cartridge	320
Ball	
Ball-tracer	
Incendiary-T	
AP	

SYSTEM

Alternative Designations: INA
Date of Introduction: Late 1970s
Proliferation: Widespread

Description:
Weight (kg):
  Loaded (with magazine): 5.0
  Empty (w/o magazine): 4.6
Length (mm):
  Overall: 1.07 m
  Barrel: 590 mm (including flash suppresser)
Rate of Fire (rd/min):
  Cyclic: 600
  Practical:
    Automatic: 150
    Semiautomatic: 50
Operation: Gas
Feed: 40-rd detachable box magazine (30-rd used by AK-74 is interchangeable)
Fire Mode: Selective, automatic or semi-automatic

SIGHTS

Name: INA
Type: Fore, cylindrical post; rear, tangent leaf with U-notch; adjustable to 1,000 m
Magnification: None
Night Sights Available: Yes. 1LH51 night sight

VARIANTS

RPKS-74: Folding stock

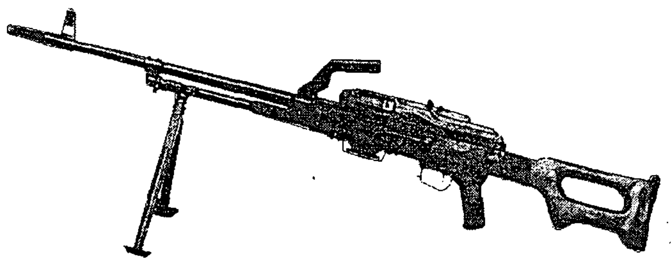
AMMUNITION

Name: 7N6
Caliber/length: 5.45x39-mm
Type: Ball
Range (m):
  Effective: 800
  Maximum: 1,000
  Armor Penetration: INA
  Muzzle Velocity (m/s): 960

Name: 7N10
Caliber/length: 5.45x39-mm
Type: AP
Range (m):
  Effective: 800
Armor Penetration (mm): 16 @ 100 m 80% of time
Muzzle Velocity (m/s): 960

NOTES
The RPK-74 is the machinegun version of the AK-74, firing the same ammunition. Instead of the prominent muzzle brake used on the AK-74, the machinegun is longer than that normally used with the AK-74, but the magazines are interchangeable. The RPK-74 has a bipod and is compatible with the front firing ports of BMPs. The RPK-74 is the standard squad machinegun in OPFOR infantry units. It generally replaces both the RPK and PKM 7.62-mm weapons.

Russian 7.62-mm General Purpose Machinegun PKM



Ammunition Types	Typical Combat Load
7.62-mm cartridge Ball Ball-tracer Incendiary-ranging API API-T	INA

SYSTEM

Alternative Designations: (see VARIANTS)
Date of Introduction (PKM/PKT): 1971/1968
Proliferation: Widespread

Description:
Weight (kg):
Empty (w/o magazine) (PKM/PKT) (kg): 8.4/10.66
Loaded (with magazine): Varies with magazine
Ammo box (only) with 100/200-rd belt (kg): 3.9/8.0
Tripod (lightweight) (kg): 4.75
Length (mm):
Overall (PKM/PKT): 1,160/1,080
On tripod (PKS): 1,267
Barrel: 658
Barrel Change: Yes
Mount Type: Pintle, coaxial, bipod or tripod (Stepanov)
Mounted On: (see VARIANTS)
Rate of Fire (rd/min):
Cyclic: 650
Practical: 250
Fire Mode: Automatic
Operation: Gas
Feed: Belt, 100-rd belt carried in a box fastened to the right side of the receiver. 25-rd belts can be joined in several combination lengths (100/200/250)

SIGHTS

Name: INA
Type: Open iron sights
Sighting range (PKM/PKT) (m): 1,500/2,000
Magnification: None
Night Sights Available: Yes

VARIANTS

PKM: Squad machinegun
PKT: Tank-mounted coaxial, lacks stock, sights, bipod, has solenoid electric trigger, longer heavier barrel.
PKS: Lightweight tripod-mounted infantry weapon
PKMS: Lightweight tripod-mounted variant of the PKS
PKB (PKBM): Pintle-mounted on APCs, SP guns, BRDM, BTRs, has butterfly trigger rather than solenoid, double space grips, and front and rear sights

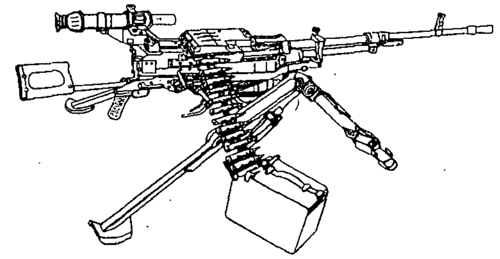
AMMUNITION

Name: INA
Caliber and Length: 7.62x54-mm rimmed
Type: Ball
Max Range (PKM/PKT) (m): 3,800/4,000
Practical Range (PKM/PKT) (m):
Day: 1,000/2,000
Night: 300/INA
Armor Penetration @ 0° obliquity @ 500 range (mm): 8
Muzzle Velocity (PKM/PKT) (m/s): 825/855

NOTES

The 7.62-mm general-purpose machinegun (PKM) is a gas-operated, belt-fed, sustained-fire weapon. The basic PKM is bipod-mounted but can also fit in vehicle firing ports. It is constructed partly of stamped metal and partly of forged steel. Compared to the US M-60, the PK-series machineguns are easier to handle during firing, easier to care for, and lighter. The 7.62x54R is a more powerful cartridge than the US with a slightly shorter effective range.

Russian 12.7-mm Heavy Machinegun NSV/NSV-T



Ammunition Types	Typical Combat Load
12.7-mm cartridge API (B-32) API-T (BZT-44) HEI	300

SYSTEM

Alternative Designations: NSVS (tripod-stand mounted), Utyos
Date of Introduction: Early 1970s
Proliferation: Widespread

Description:
Weight (kg):
  Total System (w/6T7): 43
  Empty: 25
  Loaded: INA
  Tripod (6T7 tripod): 16
Length (mm):
  Overall: 1,560
  On 6T7 Tripod: 1,900
Width (on 6T7 tripod) (mm): 860
Height (on 6T7 tripod) (mm): 380
Barrel Life (rds): 5,000
Barrel Change Time (sec): 5
Barrel Weight (kg): 9.2
Mount Type: 6T7 (infantry) tripod or 6U6 (w/seat) universal tripod
Mounted On: (see VARIANTS)
Traverse (°): 360
Elevation (°): -5 to +75
Rate of Fire (rd/min):
  Cyclic: 680-800
  Practical: 100
Fire Mode: Automatic; short bursts (four to six) or long bursts (10 to 15) or continuously
Operation: Gas
Feed: Left or right from metal link belt from 50-rd boxes

SIGHTS

Name: INA
Type: Metallic sights, (tangent leaf rear and folding front post)
Sight Range (m): 2,000
Name: 10P50 Optical
Type: Day optical sight
Magnification: 3-6x
Name: 1PN52-1
Type: Night sight
Magnification: 5.3x
Name: 10P80 (used w/ 6U6 mount)
Type: AA collimating sight (aircraft speed to 300 km/h)
Name: 10P81 (used w/ 6U6 mount)
Type: Ground target sight
Name: K10-T (on NSVT for T-72/T-80)
Type: Reflex AA sight

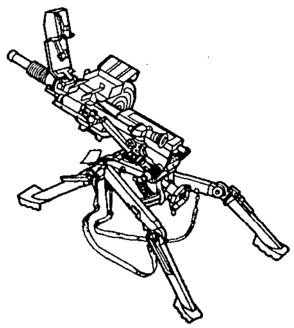
VARIANTS

NSVT: Tank-mounted, (see NOTES)

AMMUNITION
Name: B-32
Caliber and Length: 12.7x108-mm
Type: Armor Piercing Incendiary
  Max Range (ground) (m): 7,850
  Effective Range (m):
    AA: 1,000
    Ground: 2,000
    Armor: 800
    Night (w/1PN52-1): 1,000
  Armor Penetration @ 0° obliquity @ 500/1,000m range (mm): 20/13.2
  Muzzle Velocity (m/s): 860

NOTES
A tripod-mount (6T7) version is available for infantry use in a ground role. However, the NSVT appears more commonly mounted on the turrets of tanks as an antiaircraft machinegun. On the T-72 and the T-80, it has a rotating mount and can be fired from within the tank. The tank commander employs the K10-T reflex sight to engage aircraft. On the T-72/T-80 mount he engages ground targets with metallic sights on the gun itself. The T-64 tank mounts a modified version with a fixed mount on the commander's cupola. It fires by means of an electrical solenoid when the tank is buttoned up. An optic serves this purpose. Instead of the normal 50-round ammunition belt container, the NSVT on the T-64 may use a larger belt container holding 200 rounds.

Russian 30-mm Automatic Grenade Launcher AGS-17



Ammunition Types	Typical Combat Load (Dismounted)
30-mm grenade Frag-HE	87

SYSTEM

Alternative Designations: Plamya (Flame)
Date of Introduction: 1974
Proliferation: At least 12 countries

Description:
Crew: 3 (see NOTES)
Weight (kg):
  Empty (without magazine): 30.71
  Loaded (with magazine): 45.05
  Launcher: 17.86
  Sight: .99
  Tripod: 11.86
  Magazine (loaded): 14.34
Length (m): 1.28
Height (m): INA
Width (m): INA
Tripod Name: SAG-17
Mounts: Tripod, vehicle, or helicopter
Traverse (°): 30 total
Elevation (°): +7 to +87
Service Life of Barrel (rds): 6,000
Barrel Change Time: Quick disconnect
Rate of Fire (rd/min):
  Practical: 60-100
  Cyclic: 100-400 Adjustable with a thumb safety. May be fired single shot or in short (≤ 5 rds) or long (6-10 rds) bursts.
Operation: Blowback
Feed: Drum magazine containing 29 round belt.
Fire Mode: Selective, automatic and semi-automatic
Loader Type: Manual

SIGHTS

Name: PAG-17
Type: Illuminated day optical sight
Sighting Range (m): 1,700
Magnification: 2.7x
Location: Left rear of launcher
Night Sights Available: Yes

VARIANTS

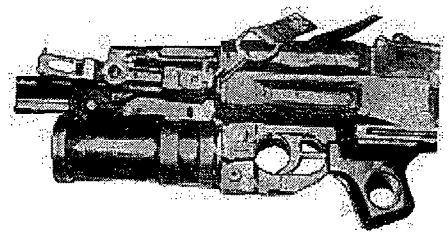
AG-17: Vehicle mounted.
AG-17A: Helicopter mounted, electric trigger, rate of fire increased to 420-500 rd/min, 300 rd belt.
TKB-722K AGL: Lighter version and possibly the follow-on to the AGS-17, shoots the same ammunition as the AGS-17

AMMUNITION

Name: VOG-17A, VOG-17M (self-destruct)
Caliber/length: 30x132.8-mm
Type: Frag-HE
Range (m)
  Direct Fire Range (m): 700
  Effective (m): 1,200
  Min Range (m): 50
  Max Indirect Range (m): 1,730
Armor Penetration: Lightly armored vehicles.
Accuracy @ 400 m:
  Distance: 4.3 m
  Deflection: .2 m
Casualty Radius (m): 15 (90% at 7 m)
Complete Round Weight (grams): 350
Grenade Weight (grams): 280
Warhead Explosive Weight (grams): 36
Muzzle Velocity (m/s): 185
Fuze Type: Impact, activates after 25 spins.

NOTES
The AGS-17 provides the infantry with an area suppressive capability. One AGL can create a damage zone 15 meters wide. The fire from an AGL platoon covers a sector approximately 90 m across. Although primarily intended for use against personnel, it has a limited capability to engage lightly armored vehicles. The crew consists of a gunner and two riflemen-assistant gunners, and may have an additional ammunition bearer. For ground transport the AGS-17 breaks down into four parts: launcher, sight, tripod, and magazine. When dismounted the gunner carries the sight and launcher, the first assistant carries the tripod and a magazine, and the second assistant carries two additional magazines. It is very accurate in the semiautomatic mode and is quite effective in area coverage in the automatic mode. The 50-meter increments in the range table atop the receiver indicate accuracy against point targets. The AGS-17 is normally organized in a platoon consisting of 6 launchers, carried in pairs in three armored vehicles (they can also be carried in trucks, or by individuals). The AGS-17 is capable of mounting night vision sights.

Russian 40-mm Under-Barrel Grenade Launcher GP-30



Ammunition Types	Typical Combat Load
40-mm grenade	10
Frag-HE (impact)	
Frag-HE (bounding)	
Smoke	

SYSTEM

Alternative Designations: BG-15 Mukha; GP-25 Koster, GP-30 Obuvka
Date of Introduction: 1980
Proliferation: Widespread

Description:
Weight (kg):
Loaded: 1.79
Empty: 1.5
Length (mm):
Overall: 323
Barrel: 205
Rate of Fire (rd/min): 4-5
Operation: N/A
Feed: Muzzle-loaded
Fire Mode: Single-shot
Accuracy @ 400 m:
Distance: 6.7 m
Deflection: 3 m
Components: Barrel (w/ mounting bracket and sight), trigger assembly

SIGHTS

Name: N/A
Type: Front post and rear open U-notched
Location: Left side of mounting bracket
Sighting Range (m): Graduated out to 400

VARIANTS

BG-15, GP-25: (see NOTES)

AMMUNITION

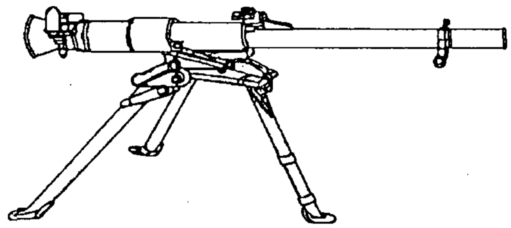
Name: VOG-25
Caliber/length: 40x102-mm
Type: Frag-HE with impact fuze
Weight (kg):
Round: .250
Exposive: .048
Range (m):
Maximum: 400
Minimum: 10–40 (arms itself)
Casualty Radius (m): 6; (90% @ 10)
Self-destruct Time (sec): 14–19
Muzzle Velocity (m/s): 76

Name: VOG-25P
Caliber/length: 40x122-mm
Type: Bounding Frag-HE, explodes .5 to 1.5 m from impact
Weight (kg):
Round: .278
Exposive: .037
Range (m):
Maximum: 400
Minimum: 10 – 40 (arms itself)
Casualty Radius (m): 6; 90% @ 10
Self-destruct Time (sec): 14 –19
Muzzle Velocity (m/s): 75

Name: GRD-40
Caliber/length: 40x150-mm
Type: Smoke
Effective Against: Visual and infrared
Weight (g): 260
Smoke Screening Range (m): 50, 100, 200
Smoke Screen Dispersion (m):
1 sec............10x10x10
2 sec............20x20x20
3 sec............25x25x25
Smoke Screen Duration @ wind speed of 3-5 m/s: At least 60 sec
Muzzle Velocity (m/s): 70-75

NOTES
The GP-30 Obuvka is a widely proliferated, muzzle-loaded, single-shot, detachable, under-barrel grenade launcher. The BG-15, GP-25 and the GP-30 are all basically the same weapon. Variants can be mounted on all models of Kalashnikov assault rifles. The rifleman can fire the launcher only when the complete weapon is attached to the assault rifle.

Russian 73-mm Recoilless Gun SPG-9



Ammunition Types	Typical Combat Load
73-mm recoilless gun RA HEAT RA HE	INA

SYSTEM

Alternative Designations: INA
Date of Introduction: 1970
Proliferation: Widespread

Description:
Crew: 3
Caliber (mm): 73
Weight (kg):
Firing Position: 47.5
Travel Position: 47.5
Tripod: 12
Length (travel) (m): 2.11
Width (travel) (m): .99
Height (travel) (m): .80
Rifling: None
Breech Mechanism Type: Interrupted screw
Feed: Breech load
Traverse (°): 30 total
Elevation (°): -3 to +7
Rate of Fire (rd/min): 6
Emplacement/displacement time (min): 1
Fire From Inside Building: No

SIGHTS

Name: PGO-9
Type: Optical and iron
Magnification: Optical 4x, 10° field of view
Location: Left side
Sighting Range (m): 1,300
Night Sights Available: IR and passive night, PGN-9

VARIANTS

SPG-9D: Airborne version with detachable wheels

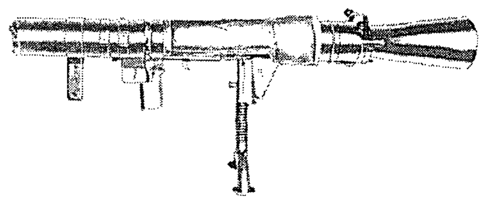
AMMUNITION

Range (m):
Maximum Effective:
HEAT: 1,000
HE: 1,300
Minimum: INA
Armor Penetration (mm) @ 1,000 m: 400 (HEAT any range)
Casualty Radius (m): INA
Length (mm): 1,000
Complete Round Weight (kg):
Rocket-Assisted HEAT: 3.5
Rocket-Assisted HE: 4
Muzzle Velocity (m/s): 435
Max Velocity w/rocket assist (m/s): 700

NOTES

The SPG-9 is a recoilless, smooth-bore, single-shot antitank weapon that fires both antiarmor and antipersonnel ammunition. Several generations of night vision equipment are available for the SPG-9. It is manportable, but a truck or APC normally carries it. It must be dismounted and placed on its tripod for firing.

Swedish 84-mm Recoilless Rifle Carl Gustaf M2



	Ammunition Types	Typical Combat Load
	84-mm round HEAT (tandem) HEDP HEAT HE Smoke Illumination	INA

SYSTEM

Alternative Designations: INA
Date of Introduction: INA
Proliferation: At least 20 countries
Description:
Crew: 1 or 2 (see NOTES)
Caliber (mm): 84
Weight (kg):
  Mount: .8
  M2: 14.2
  M3: 8.5
Length (mm): 1,065
Rifling: 24 lands/progressive twist
Breech Mechanism Type: Hinged
Rate of Fire (rd/min): 6
Fire From Inside Building: INA

SIGHTS

Name: INA
Type: Iron and telescoped
Magnification: 3x
Location: Left side
Weight (kg): 1
Used With Range Finders: Yes, laser
Night Sights Available: May be used with Generation III Image Intensification Systems.

VARIANTS
M3: Lightweight version of the M2

AMMUNITION

Name: FFV 751
Type: HEAT (tandem)
Range (m):
  Effective: 500
  Minimum: INA
  Moving: INA
Penetration:
  Armor (mm): +500
Weight (kg): 4

Name: FFV 502
Type: HEDP (with dual mode fuze)
Range (m):
  Effective (personnel in open): 1,000
  Effective (stationary): 500
  Moving: 300
  Arming Range: 15-40
Penetration:
  Armor (mm): +150
Weight (kg): 3.3
Muzzle Velocity (m/s): 230

Name: FFV 551
Type: HEAT
Range (m):
  Effective: 700
  Arming Range: 5-15
Penetration:
  Armor (mm): 400
Weight (kg): 3.2
Muzzle Velocity (m/s): 255

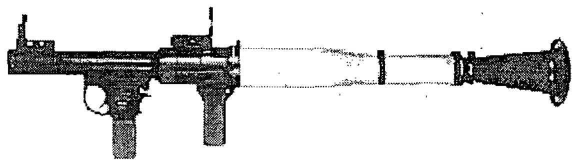
Name: FFV 441B
Type: HE
Range (m):
  Effective (unprotected troops, soft-skinned vehicles): 1,100
  Arming Range: 20-70
Casualty Radius (m): INA
Weight (kg): 3.1
Muzzle Velocity(m/s): 240

Name: FFV 469B
Type: Smoke
Range (m):
  Effective: Up to 1,300
Weight (kg): 3.1
Muzzle Velocity (m/s): 240

Name: FFV 545
Type: Illumination
Range (m):
  Practical: 300-2,100
  Burning Time (sec): 30
  Illuminated Area, dia: 400-500
  Candle Power: 650,000 cd
Weight (kg): 3.1
Muzzle Velocity (m/s): 260

NOTES
The 84-mm Carl Gustaf recoilless rifle is a one-man portable, direct-fire, single-shot, breech-loading weapon. Several versions of the Carl Gustaf are produced outside Sweden; however, the ammunition is interchangeable among the variants. While the weapon can be operated by one person it is better to have two—one to fire the gun, and the other to carry and load the ammunition. In addition to its antitank role, the weapon can be used as part of an illumination plan, to provide smoke, or for bunker busting.

Russian 40-mm Antitank Grenade Launcher RPG-7V

	Ammunition Types	Typical Combat Load
	**40-mm grenade** PG-7V PG-7VM PG-7VS PG-7VL PG-7VR TBG-7V OG-7V OG-7VM	5

SYSTEM

Alternative Designations: INA
Date of Introduction: 1962
Proliferation: At least 40 countries

Description:
Crew: 2
Caliber (launcher) (mm): 40
Weight (kg):
  Empty: 7.9
  Loaded: Varies with grenade
Length (mm): 950
Rate of Fire (rd/min): 6
Fire From Inside Building: No
Grenade Components: Warhead, rocket motor, tail assembly

SIGHTS

Name: PGO-7
Type: Optical w/II
Magnification: 2.7x, 13° field of view
Location: Top of launcher/sight-left side
Sighting Range (m): 500
Night Sights Available: Yes, NSP-3, NSP-2 (IR), NSPU, PGN-1 (II), 1PN58 (II)

VARIANTS
RPG-7D, RPG-7DV1: Folding variants used by airborne troops

AMMUNITION

Name: PG-7V
Caliber (mm): 85
Type: HEAT
Range (m):
  Effective: 500
  Minimum: INA
  Moving: 300
Penetration:
  Armor (mm): 330
Length (mm): INA
Weight (kg): 2.2

Name: PG-7VM
Caliber (mm): 70.5
Type: INA
Range (m):
  Effective: 500
  Minimum: INA
Penetration:
  Armor (mm): 330
Muzzle Velocity (m/s): 140
Length (mm): 950
Weight (kg): 2

Name: PG-7VS
Caliber (mm): 72
Type: INA
Range (m):
  Effective: 500
  Minimum: INA
Penetration:
  Armor (mm): INA
  Brick (m): + 1.5
  Reinforced concrete (m): + 1
Casualty Radius (m): INA
Muzzle Velocity (m/s): INA
Length (mm): INA
Weight (kg): 2

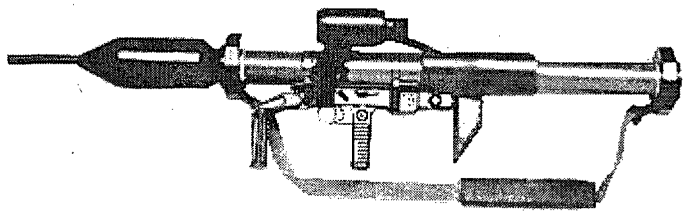
Name: PG-7VL
Caliber (mm): 93
Type: INA
Range (m):
  Effective: 300
  Minimum: INA
Penetration:
  Armor (mm): 600
  Brick (m): 1.7
  Reinforced concrete (m): + 1.1
Muzzle Velocity (m/s): 112
Length (mm): 980
Weight (kg): 2.6

NOTES
The RPG-7V is a recoilless, shoulder-fired, muzzle-loaded, reloadable, antitank grenade launcher. It fires a variety of rocket-assisted grenades from a 40-mm smoothbore launcher tube. It is the standard squad antitank weapon in use by the OPFOR. The RPG-7V is light enough to be carried and fired by one person. However, an assistant grenadier normally deploys to the left of the gunner to protect him from small arms fire. The RPG-7V requires a well-trained gunner to estimate ranges and lead distances for moving targets. Crosswinds as low as 7 miles per hour can complicate the gunner's estimate and reduce first-round hit probability to 50% at ranges beyond 180 meters.

Russian Antitank Grenade Launcher RPG-7V continued

Name: PG-7VR (uses RPG-7V1 launcher sights) Caliber (mm): 105 Type: Tandem Range (m): Effective: 200 Minimum: INA Sighting Range: INA Penetration: Armor (mm): +750 (all armor including reactive armor) Brick (m): 2 Reinforced concrete (m): +1.5 Muzzle Velocity (m/s): INA Length (mm): 1,306 Weight (kg): 4.5 **Name**: TBG-7V (uses RPG-7V1 launcher sights) Caliber (mm): 105 Type: Thermobaric (similar to RPO-A warhead) Range (m): Effective: 200 Sighting Range: 800 Penetration: Armor (mm): INA Brick (m): +1.5 Reinforced concrete (m): + 1.5 Casualty Radius (m): INA Muzzle Velocity (m/s): INA Length (mm): INA Weight (kg): 4.5	**Name**: OG-7V Caliber (mm): 40 Type: Frag-HE Range (m): Effective: 950 Casualty Radius (m): INA Muzzle Velocity (m/s): 152 Length (mm): 569 Weight (kg): 1.7 **Name**: OG-7VM Caliber (mm): 40 Type: Frag-HE Range (m): Effective: 1,000 Casualty Radius (m): INA Muzzle Velocity (m/s): 145 Length (mm): 595 Weight (kg): 1.7

German 60-mm Antitank Grenade Launcher Panzerfaust-3



Ammunition Types	Typical Combat Load
60-mm grenade HEAT Multipurpose-FRAG BASTEG Illumination Smoke Practice	INA

SYSTEM

Alternative Designations: Balliste, Pzf 3
Date of Introduction: 1990
Proliferation: At least eight countries

Description:
Crew: 1
Caliber (mm):
  Launch Tube: 60
  Warhead: 110
Weight (kg): 12
Length (mm):
  Firing Position: 1,200
  Travel Position: 1,200
Rifling: None
Breech Mechanism Type: N/A
Rate of Fire (rd/min): 5
Fire From Inside Building: Yes

SIGHTS

Name: INA
Type: Optical
Magnification: INA
Location: Left side
Used With Range Finders: Yes
Night Sights Available: Yes

VARIANTS

Panzerfaust 3-T 600: Simrad IS2000 laser gun sight with range of moving targets out to 600 m. Can be fitted with Simrad KN205F night sights.

AMMUNITION

Name: INA
Caliber (mm): 110
Type: HEAT
Range (m):
  Effective (moving): 300
  Effective (stationary): 500
Penetration:
  Armor (mm): + 700
Weight (kg): 3.8
Muzzle Velocity(m/s): 170
Flight Velocity(m/s): 250
Time of Flight to 300 m (sec): 1.3

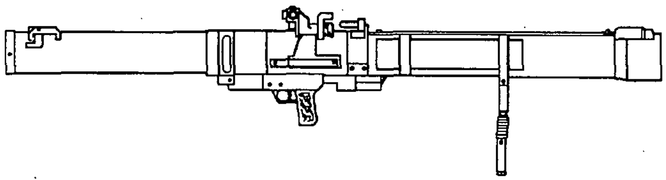
Name: BASTEG (Barricade and Street Encounter Grenade)
Caliber (mm): 110
Type: Shaped-charge w/stand-off fuze
Range (m): INA
Penetration:
  Concrete (mm): INA
Weight (kg): INA
Muzzle Velocity(m/s): INA
Flight Velocity(m/s): INA
Time of Flight to 300 m (sec): INA

OFF-ROUTE MINE SYSTEM

Target Speed range (km/h): 30-60
Effective Range (m): 150
Operational Time (days): 40
Acquisition: Targets detected by acoustic sensor which activates the infra-red sensor.
Sensors:
IR Sensor: Passive, two-color
IR Optics: Double parabolic, off-axis
Acoustic: Capacitative microphone.

NOTES
The Panzerfaust 3 is a compact, lightweight, shoulder-fired, unguided antitank weapon. It consists of a disposable cartridge with a 110-mm warhead and reusable firing and sighting device. The Panzerfaust can be adapted to serve as an off-route mine.

Russian 105-mm Antitank Grenade Launcher RPG-29



Ammunition Types	Typical Combat Load
105-mm grenade HEAT (tandem)	INA

SYSTEM

Alternative Designations: Vampir
Date of Introduction: Late 1980s
Proliferation: Former Soviet Union

Description:
Crew: 2
Caliber (tube) (mm): 105
Weight (kg): 11.5
Length (mm): 1,000
Life of Tube/barrel: 300
Rate of Fire (rd/min): INA
Fire From Inside Building: INA
Maximum Target Speed (km/h): INA
Emplacement/displacement time (min): (see NOTES)

SIGHTS

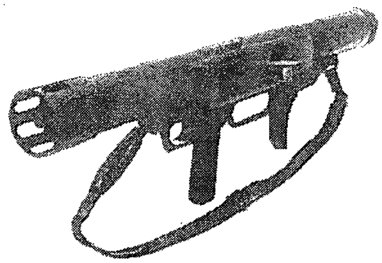
Name: INA
Type: Iron, optical, and night
Magnification: INA
Location: Left side
Sighting Range (m): 450
Night Sights Available: Yes, INA

VARIANTS (see NOTES)

AMMUNITION
Name: PG-29V
Caliber (warhead): 105
Type: HEAT (tandem)
Range (m):
Effective: 500
Minimum: INA
Penetration (m):
Armor: +750, (650 behind ERA)
Concrete and brick: +1.5
Casualty Radius (m): INA
Length (mm): INA .
Complete Round Weight (kg): 6.7
Muzzle Velocity (m/s): 280

NOTES
For ease of transportation the RPG-29 can be broken down into two parts which one soldier can carry. It can be made ready to fire within a few seconds. A folding bipod is provided to assist aiming during prone firing. An unnamed variant has a tripod mount and guidance and control system. The guidance and control system of the mounted variant includes an optical sight, laser rangefinder and ballistic data computer for firing on moving targets. This increases the effective range of the mounted system to 800 m against a stationary target with a hit probability of 80%.

German 67-mm Disposable Antitank Grenade Launcher Armbrust

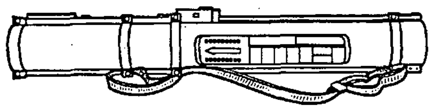
	Ammunition Type	Typical Combat Load
	**67-mm grenade** HEAT	**INA**

SYSTEM **Alternative Designations:** Crossbow **Date of Introduction:** INA **Proliferation:** At least seven countries **Description:** Crew: 1 Caliber (mm): 67 Weight (kg): 6.3 Length (mm): 850 Rifling: None Breech Mechanism Type: N/A Rate of Fire (rd/min): N/A (disposable) Fire From Inside Building: Yes (see NOTES) **SIGHTS** **Name:** N/A Type: Reflex Magnification: None Location: Left side Sighting Range (m): INA **Night Sights Available:** INA	**VARIANTS** (INA) **AMMUNITION** **Name**: INA Type: HEAT Range (m): Maximum: 1,500 Effective AT: 300 Flight Time (sec) @ 300 m: 1.5 Penetration: Armor (mm): 300 Reinforced Concrete (m): INA Muzzle Velocity(m/s): 210

NOTES

The Armbrust is a preloaded, disposable, shoulder-fired antitank weapon. It has a low signature and low IR detectability and can be safely fired from small enclosed rooms. The muzzle does not emit smoke or blast and no flash can be seen from the rear. Only .8 m clearance is required between the rear of the weapon and the wall. It is quieter than a pistol shot. The entire weapon is considered a round of ammunition and the launcher is thrown away once the weapon is fired. Manufactured by Singapore.

Russian 72-mm Disposable Antitank Grenade Launcher RPG-22



Ammunition Types	Typical Combat Load
72-mm grenade HEAT	**INA**

SYSTEM

Alternative Designations: INA
Date of Introduction: 1985
Proliferation: At least three countries

Description:
Crew: 1
Caliber (mm): 72
Weight (kg): 2.8
Length (mm):
  Firing Position: 850
  Travel Position: 750
Rifling: None
Breech Mechanism Type: N/A
Rate of Fire (rd/min): N/A (disposable)
Fire From Inside Building: No, backblast out to 30 m behind the weapon.

SIGHTS

Name: INA
Type: Iron, calibrated for 50, 150, 200 m
Magnification: None
Location: Top of launcher
Sighting Range (m): 250
Night Sights Available: No

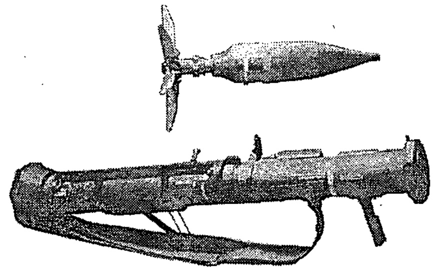
VARIANTS (None)

AMMUNITION (see NOTES)

Name: INA
Caliber (mm): 72
Type: HEAT
Range (m):
  Effective: 250
  Arming Range: INA
Penetration:
  Armor (mm): 390
  Brick (m): 1.2
  Reinforced Concrete (m): 1
Muzzle Velocity(m/s):
  Initial: 133
  Maximum: 300
Length (mm): 618
Weight (kg): 1.48

NOTES
The RPG-22 is a lightweight, shoulder-fired, preloaded, disposable antiarmor weapon intended for firing one round, after which the tube is discarded. It is basically a scaled-up version of the RPG-18 (similar to the US LAW) and has no dedicated grenadier; however, all soldiers train to use the squad-level disposable weapon.

Swedish 84-mm Disposable Light Antitank Weapon AT4



Ammunition Types	Typical Combat Load
84-mm round HEDP HEAT	INA

SYSTEM

Alternative Designations: US M136, Bofors AT 4, FFV AT4
Date of Introduction: INA
Proliferation: At least seven countries

Description:
Crew: 1
Caliber (mm): 84
Weight (kg): 6
Length (mm):
  Firing Position: 1,000
  Travel Position: 1,000
Rate of Fire (rd/min): N/A (disposable)
Fire From Inside Building: See AT4 CS

SIGHTS

Name: INA
Type: Popup, preset to 200 m
Location: Top left
Night Sights Available: Yes, INA

VARIANTS (see NOTES)

LMAW: Light Multipurpose Assault Weapon, uses HEDP
AT4 CS: Confined space
AT4 HP: High penetration

AMMUNITION

Name: AT4 HEAT
Caliber (mm): 84
Type: HEAT
Range (m):
  Effective: 300
  Arming Range: INA
Penetration:
  Armor (mm): 420
Weight (kg): 6.7
Muzzle Velocity(m/s): 285

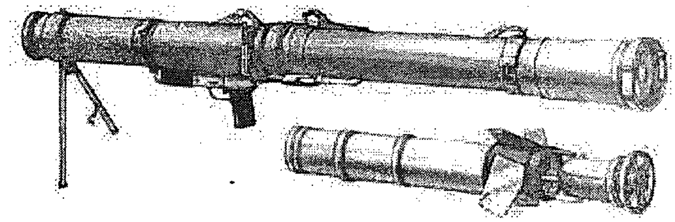
Name: LMAW (see VARIANTS)
Caliber (mm): 84
Type: HEDP, modified Carl Gustaf HEPD FFV 502 (with dual mode fuze)
Range (m):
  Effective: 300
  Arming Range: INA
Penetration:
  Armor (mm): 150
  Concrete (m): INA
Casualty Radius (m): INA
Muzzle Velocity (m/s): 235

Name: AT4 CS (confined space) can fire from confined spaces as small as 22.5 m^3
Caliber (mm): 84
Type: HEAT or HEDP (LMAW) warheads
Range (m):
  Effective: INA
  Arming Range: INA
Penetration:
  Armor (mm): INA.
Weight (kg): INA
Muzzle Velocity(m/s): INA

Name: AT4 HP (high penetration)
Caliber (mm): 84
Type: HEAT
Range (m):
  Effective: INA
  Arming Range: INA
Penetration:
  Armor (mm): 600
Weight (kg): Less than 7
Muzzle Velocity(m/s): 290

NOTES
The AT4 is a lightweight, preloaded, disposable antiarmor weapon intended for firing one round, after which the tube is discarded. All AT4 systems share the same launcher but may contain different preloaded munitions. The variant selected depends on the intended use. The AT4's average recoil is comparable to the M16 rifle.

Russian Infantry Rocket Flame Weapon RPO



Ammunition Types	Typical Combat Load
Rocket Rocket-propelled encapsulated napalm projectile.	2

SYSTEM
Alternative Designations: Rys (Lynx)
Date of Introduction: Late 1970s'
Proliferation: FSU

Description:
Crew: 1
Weight (kg):
Empty: 3.5
Pack (launcher and two rounds): 22
Length (ready to fire) (m): 1.44
Rate of Fire (rockets/min): 1
Reaction Time-Travel to Fire (sec): 60
Fire From Inside Building: INA
Tube Life: 100 rounds
Launcher Components: Firing tube, firing mechanism, mechanical sights, collapsing bipod and sling.

PERFORMANCE

Range (m):
Effective: 190
Maximum: 400
Minimum: INA
Accuracy: INA
Muzzle Velocity (m/s): INA

SIGHTS

Name: N/A
Type: Open metal, front and rear
Location: Left side, rear is on-line with rear of grip
Magnification: None
Night Sights Available: INA

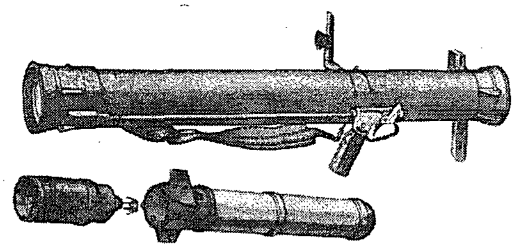
VARIANTS (None)

AMMUNITION

Name: RPO
Type: Incendiary
Warhead Incendiary Fill (liters): 4
Weight of Incendiary in Warhead (kg): 4
Type of Incendiary: Pyrogel
Burn Temperature (°C): 800-1,000
Caliber (mm): 122
Casualty Radius: Fire envelope 10-40 m deep in the direction of the shot with a spray width of 3-4 m.
Components: Container, warhead canister, propulsion unit

NOTES
The RPO is a combat-tested, shoulder-fired reusable weapon that fires a rocket-propelled encapsulated napalm warhead. It was designed to replace the LPO-50. The RPO is carried in two parts that must be connected to fire. Squeezing the trigger ignites the rocket with an electric spark. Part of the propellant gas enters the container and pushes the canister, kindling the igniter which in turn, ignites the incendiary mixture. The napalm in the RPO ignites at the initial stage of the flight and upon impact burning pieces are scattered all over the target. Although still in use by the OPFOR Flamethrower Bn (Encapsulated) at Corps or Army level (and other armies), the RPO has generally been replaced by the Infantry Rocket Flame Weapon RPO-A Series (RPO-A/D/Z).

Russian Infantry Rocket Flame Weapon RPO-A Series (RPO-A/Z/D)



Ammunition Types	Typical Combat Load
Rocket RPO-A: Thermobaric-flammable mixture RPO-Z: Incendiary RPO-D: Smoke	2

SYSTEM

Alternative Designations: Shmel (Bumblebee)
Date of Introduction: 1984
Proliferation: Widespread

Description:
Crew: 1
Caliber (mm): 93
Number of Weapons in a Package: 2
Weight of Package (kg): 12
Total weapon (1) weight (kg): 11
Length (mm): 920
Rate of Fire (rockets/min): 2
Reaction Time-Travel to Fire (sec): 30
Fire From Inside Building: Yes. It can be fired in enclosures of 60 m^3 or greater or with a barrier behind the weapon.
Components: Container, ejection motor, warhead.

PERFORMANCE

Range (m):
Direct Fire: 200
With Optical Sight: 850
Effective: 600
Minimum: 20
Indirect Fire: 1,000
Accuracy @ 200 m: .5 m^2
Muzzle Velocity (m/s): 125

SIGHTS

Name: OPO-1
Type: Optical calibrated to 600 m
Location: Left, next to grip
Magnification: None
Night Sights Available: INA

VARIANTS (None)

AMMUNITION

Name: RPO-A
Type: Thermobaric
Casualty Radius (m): 50 (personnel in open)
Lightly armored materiel kill probability at 400 m: 0.7
Burn Temperature (°C): 800+
Warhead Explosive Type: Trotyl equivalent (kg) -2
Warhead Mixture Weight (kg): 2.1

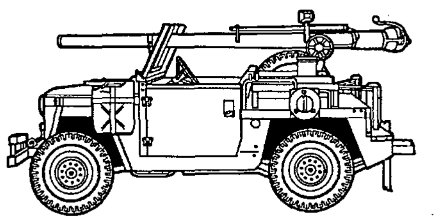
Name: RPO-Z
Type: Incendiary
Warhead Mixture Weight (kg): 2.5

Name: RPO-D
Warhead Weight (kg): 2.3
Smoke-Incendiary Type: Based on red phosphorous.
Smokescreen:
Time of Formation (min): 2
Length (m): 55 to 90
Depth (m): INA
Height (m): INA
Duration (min): 3 to 5
Effective Against: Visual and infrared

NOTES

Designed as a follow-on to the RPO, the RPO-A, -Z, and -D are one-shot, disposable, shoulder-fired, combat tested (Afghanistan, Tajikistan, Chechnya), flame weapons. They are reliable and can be ready to fire within 30 seconds. Any soldier, infantryman, or paratrooper can use this close-combat weapon with minimal instruction. The RPO-A comprises three basic components: container, ejection motor, and case which is filled, depending on its purpose, with thermobaric (enhanced blast explosive), smoke or incendiary rockets. At any range the blast effects of the thermobaric munitions are much more serious than the thermal effects. The RPO-A is known as the infantryman's pocket artillery because the demolition effect corresponds to the 122-mm HE artillery, and 120-mm mortar projectile. The RPO series of flame weapons also serves as an extremely effective counter-sniper weapon. The armor- and mechanized -based OPFOR usually issues one RPO-A per BMP (mechanized infantry squad). They are also found in the Flamethrower Bn (Encapsulated) at Corps or Army level. One squad per infantry platoon has a RPO-A in the infantry-based OPFOR. The RPO-A series of flame weapons are issued more along the lines of ammunition rather than a weapon, therefore the BOI may vary.

United States 106-mm Recoilless Rifle M40



Ammunition Types	Typical Combat Load
106-mm recoilless gun	INA
HEAT	
HEAT-T	
HEP-T	
APERS-T	
HEAP	

SYSTEM

Alternative Designations: (see VARIANTS)
Date of Introduction: 1953
Proliferation: At least 50 countries

Description:
Crew: 2
Caliber (mm): 106
Weight (kg):
With Spotting Rifle: 130
Gun Only: 113
Length (m):
Total: 3.40
Barrel: 2.85
Width (on M79 mount) (m):
Legs Spread: 1.524
Legs Closed: .8
Height (on M79 mount) (m): 1.3
Bore: Rifled 36 grooves, rh
Breech Type: Interrupted thread
Recoil System: Vented breech
Feed: Manual
Traverse (°): 360
Elevation (°) (M79 Mount): -17/+65
Rate of Fire (rd/min): 5
Spotting Rifle: .50 cal M8C
Emplacement/displacement time (min): INA
Fire From Inside Building: No
Complete Round Weight (kg): 13
Muzzle Velocity (m/s): 570

SIGHTS

Name: INA
Type: Optical
Name: Bofors modernization package
Type: Simrad LP101 laser sight in place of the ranging gun
Magnification: INA
Location: INA
Name: Bofors modernization package
Type: Computerized LASer Sight (CLASS)
Magnification: INA
Location: INA
Night Sights Available: Yes, INA

VARIANTS

M79 Mount: Tripod, ground, or vehicle
M50 Ontos: Six-barrel mount on small tracked vehicle
PAK-66: Austrian M40 on two-wheel carriage

AMMUNITION

Name: M344A1
Type: HEAT
Range (m):
Maximum Effective: 1,350
Maximum Range: 2,745
Armor Penetration (mm): INA
Complete Round Weight (kg): 16.8
Muzzle Velocity (m/s): 503

Name: 3/A-HEAT-T (Bofors upgrade)
Type: HEAT-Tracer
Range (m):
Maximum Effective: 2,000
Armor Penetration (mm): 700 +
Complete Round Weight (kg): 14.5
Muzzle Velocity (m/s): 570

Name: M346A1
Type: HEP-T (HE plastic-tracer)
Range (m):
Maximum: 6,870
Complete Round Weight (kg): 16.95
Muzzle Velocity (m/s): 498

Name: M581
Type: APERS-T (antipersonnel-tracer) (flechette)
Fill (.5 g ea): 10,000 flechettes
Range (m):
Maximum Effective: 300
Complete Round Weight (kg): 18.73
Muzzle Velocity (m/s): 438

Name: HEAP M-DN
Type: HE antipersonnel (steel pellets)
Fill: 1,000 steel pellets
Range (m):
Maximum Effective: 1,500
Lethal Radius: 40
Complete Round Weight (kg): 16.4
Muzzle Velocity (m/s): 560

NOTES
The US M40 or M40A1 recoilless rifle is an antitank weapon. It uses a .50 cal spotting rifle mounted along the axis of the barrel to determine proper elevation for the 106-mm barrel. Upgraded systems may have the Simrad laser sight in lieu of the ranging (spotting) gun.

Chapter 2
Infantry Vehicles

Infantry vehicles can vary from general transport assets such as trucks, to specially designed ***light armored fighting vehicles (LAFVs)***. The intensity of combat on the modern battlefield requires infantry vehicles that are mobile, survivable, and lethal. Many ground forces have programs underway to field infantry LAFVs for modern requirements. Because of budgetary constraints, many ground forces continue using infantry vehicles which we might consider obsolete, but which are well suited for their environment and military role. A number of forces have aggressive upgrade programs for older systems. The U.S. Army, in its next conflict, is likely to encounter infantry forces with a mix of older and newer infantry vehicles.

CLASSIFICATION

Infantry LAFVs are generally classed as ***armored personnel carriers*** (APCs) or ***infantry fighting vehicles*** (IFVs). The lighter, less protected and less lethal system is the APC. It is intended to carry soldiers to the close combat zone, then dismount them for their commitment to the fight. An IFV is designed to fight with soldiers onboard, to carry the soldiers forward without dismounting them if possible, and to support them with direct fires if they do dismount. The plethora of upgrade options available is permitting both APCs and IFVs to become more mobile, survivable, and lethal. Thus we see APCs with IFV survivability or IFV lethality, or with both—which transforms them into IFVs. We also see IFVs with vulnerabilities which ill-suit them for their mission requirement. This chapter highlights key infantry vehicles, with an emphasis on their capabilities in mobility, survivability and lethality. Please note that on the modern battlefield, lack of a capability (swim, night sights, etc.) is in fact a vulnerability.

TRENDS

This chapter highlights infantry LAFV features in terms of mobility, survivability, and lethality. Armies have been looking at ways to balance the need for increased protection with limitations that additional armor brings, such as the need to be amphibious. One solution is to accept a lack of swim capability for a segment of up-armored IFVs, coupled with a distribution of (less armored) amphibious vehicles within the force. Other armies are looking at limited addition of applique armor or active protection systems. Several companies have developed light explosive reactive armor (ERA), which can be used on LAFVs. However, this is a less likely upgrade, because exploding armor fragments are a hazard to dismounted soldiers.

In the past, higher combat power and cost of tanks justified the wide disparity in firepower between tanks and IFVs. However, modern IFVs, when fully manned and equipped, may have equal or higher combat power and similar cost. Therefore, lethality improvements previously afforded to tanks are being added to selected IFVs. A wide variety of lethality upgrades are available for LAFVs. These include larger main weapons and antitank guided missile (ATGM) launchers, and improved fire control systems (FCS), especially night sights. The simplest but sometimes most costly upgrade is improved ammunition.

Improved secondary armaments for aerial targets permit the main weapon to focus more on heavy targets. Thus, several countries are adding remote day sights and night sights and improved ammunition for machineguns (MGs). Others are adding automatic grenade launchers to supplement MG fires.

The aerial threat to AFVs has prompted ground forces to address that threat. One response is proliferation of air defense assets, such as shoulder-fired SAMs. A more direct response which is difficult to counter, is cost-effective, and has long-term benefits for force effectiveness, is to better equip the vehicles for counterair fires. Some infantry vehicles have been fitted with high-angle-of-fire turrets (e.g., BTR-80) and antiaircraft sights (BMP-3). Improved fire control technology has led to more exotic ammunition solutions. The BMP-3 gun-launched ATGM has a higher velocity for use against helicopters. Another new development is ballistic computer-based electronically-fuzed frag-HE rounds, including forward- and side-firing rounds, which can defeat rotary-wing aircraft and ground-based antiarmor positions at stand-off range.

Infantry vehicles offer the most economical armored vehicle chassis for development of combat support and service support vehicles, including air defense vehicles, artillery, C^4, reconnaissance, etc. Noted variants offer a link to other systems described in the WEG.

This chapter provides a representative sampling of infantry vehicles in use today. The selection is not comprehensive, rather reflects APCs and IFVs currently available to the OPFOR. Within this chapter, other types of infantry vehicles are also noted. These include airborne vehicles and multipurpose transporters. Other armored transport vehicles available to infantry units are armored trucks (e.g., former Soviet BTR-152), amphibious assault vehicles (such as U.S. LVTP7), jeep-type vehicles (e.g., HMMWV), and fast-attack vehicles (based on so-called dune buggy designs). Examples of alternative vehicles will be added in later issues of the WEG. .

TECHNICAL NOTES

The following notes apply to infantry LAFVs, and to combat vehicles (in other chapters) that are used for reconnaissance, tank/assault, antitank, air defense, and artillery roles. Weapon, fire control, and munition-related narrative applies to towed and ground weapon systems.

On each equipment sheet, the top of the page provides an illustration (line drawing or photo of the system) and a summary of weapons and munitions. Note that a Typical Combat Load, when available, may be estimated. In actuality, ammunition load depends on specific country holdings, on time frame, and on scenario tactical considerations.

System and Variants sections provide basic data to assist in understanding current system status and proliferation, as well as possible upgrade options. Under Description, to assure comparability on vehicle dimensions, gun tube length is not included in those dimensions.

In the area, Automotive Performance, the figure *max off-road* denotes speed on dirt roads. The figure *average cross-country* is used for true off-road speed; for selected systems, it was measured on an approved course. Although some systems have specified radios, for many OPFOR countries, radios will be replaced to link with their military radio nets.

Protection figures for use in simulation applications must be measured by certifying agencies in accordance with specific Army standards. Figures on equipment sheets include published data provided for general information use, and may not coincide with vulnerability data developed by approved agencies. Protection options are available for upgrading systems. The wide variety of supplemental protection packages include active and passive armor, active protection systems and countermeasure systems. Although upgrades are being advertised and are technically possible, that does not mean that they are tactically sound, or that the application fits the OPFOR to be portrayed. Other options are generally available for installation; but, because their applicability has not been noted for specific systems, they were not included. Only a few countermeasure parameters were included. However, specific protection upgrades and systems are noted for selected OPFOR systems.

System lethality is determined by a variety of interrelated functions and considerations in the process of bringing destruction upon enemy forces and equipment. Lethality is addressed on the equipment sheets under the headings of Armament, Fire Control, Sights, and Main Armament Ammunition. Lethal fires can be delivered by ***direct fire***, in which weapon systems acquire and observe their targets, or by ***indirect fire***, in which weapons use remote acquisition assets to direct their fires. Note that direct-fire systems such as tanks can receive remote acquisition reports and engage targets by indirect fire; and indirect fire systems (such as artillery) can employ direct-fire sights to fire in the direct-fire mode. For the WEG, high-angle fires are not interpreted as indirect fires as long as the firing weapon uses its own sights to acquire and aim.

Factors affecting lethality, which are considered in the WEG, include: rates of fire, various ranges, accuracy and errors, acquisition/fire control capabilities, lethality effects, ammunition, and ability to engage targets on the move. Any of these technical factors, and other more subtle ones, may affect lethality in combat. Note also that various rates of fire are used, with adjusting factors, such as movement status and type of target. Generally automatic weapon use life dictates that, for more than a 3-4 second interval, the number of rounds expended will not exceed the ***practical*** rate of fire. However, maximum rate is critical against fast-closing targets, such as flying aircraft.

Range is not a fixed figure for most systems. It can be directly affected by four technical factors: gun/launcher configuration, mount (how it is fixed to the system), acquisition capability, and specific munition ballistics. Range is also related to less tangible factors, such as movement status (moving versus stationary, and movement speed), target type, elevation angle (such as for air defense weapons), visibility conditions, and terrain. Each weapon can have different ranges listed by ammunition type and model, where munitions are broken out. Generally, the range of direct-fire frag-HE rounds is greater than munitions designed for point targets, because the effects area is much greater than those of shaped-charge or kinetic-energy rounds. With fragmentation and blast effects, a near miss may be good enough to inflict severe damage. With these considerations, the WEG provides a figure called ***maximum aimed range***. This range indicates the farthest range for system-on-system aimed direct fire.

The maximum aimed range is based on a combination of tactics, techniques and procedures (TTPs), and on parameters of the technical factors noted above: gun/launcher, mount, acquisition system, and ammunition ballistics. This direct-fire range significantly exceeds the weapon's ***maximum effective range***. The maximum effective range/night denotes the effective range for a round, given available night acquisition capabilities. The TTPs also call for a "salvo range" for armored fighting vehicles, which exceeds other ranges and requires one or more volleys of a platoon against a single point target. These figures are less tangible, are based on TTP, and are not included in the WEG.

Probability of hit data is included for instructional purposes, not for use in simulations and models. Accuracy for weapons, munitions, and acquisition systems decreases with range. Antitank guided missiles are an exception; they usually have a singular probability of hit for all ranges, based on technical precision capability. Limitations, vulnerabilities, and countermeasures can affect actual performance. Several of these factors are noted on equipment pages.

Lethality performance given a hit can be measured in terms of radius of effects for fragmentation/blast effects against soft targets, and penetration distance (through steel) against hard targets. The fragmentation and blast effects of a frag-HE round mean that it is less lethal against hard targets, such as heavily armored vehicles. Another consideration is the level of destruction required. For many possible adversary forces, the critical requirement against armored vehicles is not a 100% or catastrophic kill. A mobility kill or firepower kill may be sufficient to render a system combat-ineffective, and may be counted in lethality data. The OPFOR can employ a mix of lethal and nonlethal methods. Fires of degrading (versus destructive) munitions such as smoke, mines, and radio frequency jammers can be used to suppress units and support the effort. Consult other manuals in the FM 100-60 series and other approved publications for guidance on these tactics, techniques, and procedures.

Questions and comments on data listed in this chapter should be addressed to:

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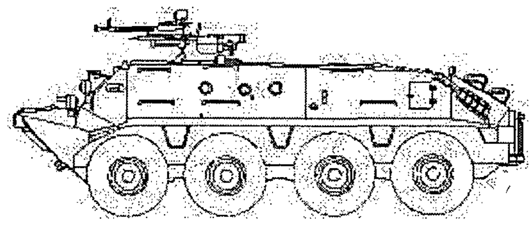
Russian Armored Personnel Carrier BTR-60PA



Weapons & Ammunition Types	Typical Combat Load
12.7-mm DShK MG	**500**
APDS, API, API-T,	250
HE-T, HEI, I-T	250
2 x 7.62-mm PKT MG	**3,000**
Lt Ball, Ball-T	
API, API-T	

SYSTEM
Alternative Designations: BTR-60-PK
Date of Introduction: 1963
Proliferation: At least 30 countries (including variants)
Description:
Crew: 2
Troop Capacity: 12
Combat Weight (mt): 10.1
Chassis Length Overall (m): 7.22
Height Overall (m): 2.06
Width Overall (m): 2.82
Ground Pressure (kg/cm^2): INA
Drive Formula: 8 x 8

Automotive Performance:
Engine Type: 2 x 180-hp Gasoline
Cruising Range (km): 500
Speed (km/h):
Max Road: 80
Max Off-Road: 60
Average Cross-Country: INA
Max Swim: 10
Fording Depths (m): Amphibious

Radio: INA

Protection:
Armor, Turret Front (mm): 7-9mm hull front (no turret)
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: Collective
Smoke Equipment: N/A

ARMAMENT
Main Armament:
Caliber, Type, Name: 12.7-mm (12.7 x 108) heavy MG, DShK
Rate of Fire (rd/min): 80-100 (practical)
Loader Type: Belt feed
Ready/Stowed Rounds: INA
Elevation (°): -10/+80
Fire on Move: Yes

Auxiliary Weapons:
Caliber, Type, Name: 2 x 7.62-mm machinegun PKT
Mount Type: Vehicle top
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,000
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, in 2-10 rd bursts

ATGM Launcher: N/A
Firing Ports: 3 on each side

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: N/A
Sights w/Magnification:
Gunner:
Day: K10-T
Field of View (°): INA
Acquisition Range (m): 1,500 (est)
Night: N/A
Commander Fire Main Gun: No

VARIANTS
A variety of armament variants for the vehicle were used, including single 7.62-mm PKT MG, or 12.7-mm MG, or no MG.

Artillery command and reconnaissance vehicles. ACRV **1V18** is a command and observation vehicle (COP). ACRV **1V19** is a fire direction center (FDC).

BTR-60PB: The most widely fielded variant has a one- man turret, a 14.5-mm KPV-T MG, a coaxial 7.62-mm MG and day/night sights.

BTR-60PBK: Company commander variant with 3 additional radios

BTR-60 PU: Armored command vehicle (ACV) variant with a 10-m mast radio antenna and front-to-rear rail antenna for mobile use

BTR-60 PU-12/ -12M: Air defense associated ACV and its upgrade

BTR-60 R-975: Forward air controller turreted variant.

MTP-2: Armored recovery vehicle

R-145BM: ACV with R-111, R-123, and R-130M radios and the distinctive Clothesline antenna

Russian Armored Personnel Carrier BTR-60PA continued

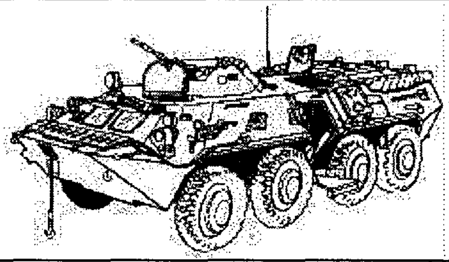
MAIN ARMAMENT AMMUNITION	Other Ammunition Types
Caliber, Type, Name: 12.7-mm, APDS Chinese, Type 54 Maximum Aimed Range (m): 1,500 Max Effective Range (m): Day: 1,500 vehicles Night: N/A Tactical AA Range: 1,600 Armor Penetration (mm): INA 12.7-mm, API/API-T Type 54 Maximum Aimed Range (m): 1,500 Max Effective Range (m): Day: 1,500 unarmored ground / 800 armored Night: N/A Tactical AA Range: 1,000 Armor Penetration (mm): INA	**Other Ammunition Types:** Incendiary-T, HE-T Type MDZ, HEI Type ZP, Russian Duplex, Russian Duplex-T

NOTES

This vehicle is a roofed variant of the BTR-60P open-hatch armored carrier. It is widely fielded in original and modified form. The APC has a top-mounted 12.7-mm MG forward of rectangular gunner's hatch. Where an additional two 7.62-mm MGs are mounted, they are right and left of the hatch. Because of space restriction, no more than one or two gunners can fit in the opening.

A notable vulnerability is that passengers have to exit the vehicle through top hatches, which makes them vulnerable to fires. Also, gunners must be at least shoulder high out of the vehicle to operate the weapons.

Russian Armored Personnel Carrier BTR-80



Weapons & Ammunition Types	Typical Combat Load
14.5-mm turret MG API, API-T	**500**
7.62-mm coax PKT MG Light ball, Ball-T, API, API-T, I-T	**2,000**

SYSTEM
Alternative Designations: GAZ-5903
Date of Introduction: 1984
Proliferation: At least 22 countries
Description:
Crew: 2
Troop Capacity: 8
Combat Weight (mt): 13.6
Chassis Length Overall (m): 7.55
Height Overall (m): 2.41
Width Overall (m): 2.95
Ground Pressure (kg/cm^2): INA
Drive Formula: 8 x 8

Automotive Performance:
Engine Type: 260-hp Diesel
Cruising Range (km): 600
Speed (km/h): Max Road: 85
Max Off-Road: 60
Average Cross-Country: 40
Max Swim: 10
Fording Depths (m): Amphibious

Radio: R-173

Protection:
Armor, Turret Front (mm): Against 12.7mm
Applique Armor (mm): Available
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: Collective
Smoke Equipment: 6 x 81-mm smoke grenade launchers

ARMAMENT
Main Armament:
Caliber, Type, Name: 14.5-mm (14.5 x 114) heavy MG, KPVT
Rate of Fire (rd/min): 150 practical
Loader Type: Belt-fed
Ready/Stowed Rounds: 50/450
Elevation (°): -4/+60
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm machinegun, PKT
Mount Type: Coax
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,000
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical/650 cyclic, 2-10 round bursts
ATGM Launcher: N/A
Firing Ports: 3 on each side

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: OU-3GA2M
Sights w/Magnification:
Gunner:
Day: 1P3-6, 1.2x/4x
Field of View (°): 49/14
Acquisition Range (m): 2,000
Night: N/A
Commander Fire Main Gun: No

VARIANTS
BTR-80A: IFV with a stabilized turret containing a 30-mm 2A42 (BMP-2) automatic gun, coax 7.62-mm MG, and TNP-3 day/night sights. This vehicle is in prototype and offered for export. The drop-in turret package is also offered for export. A BTR-80S variant includes the turret/fire control, but with 14.5-mm versus 30-mm gun.

BTR-80K: Command variant w/ added R-173, R-173P, and R-159 (portable) radios, R-174intercom, navigation aids, and an 11-m mast

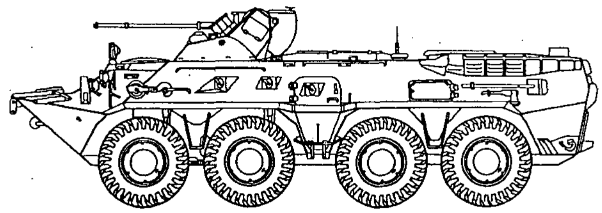
1V152: Artillery battalion command vehicle
2S23: 120-mm self-propelled combination gun (howitzer/mortar)
BREM-K: Armored recovery vehicle
RKhM-4: NBC reconnaissance vehicle
UNSh: Standardized expanded chassis for current variants, including **Kushetka-B** ACRV and BMM vehicle series.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
14.5-mm API-T
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 2,000
Night: INA
Tactical AA Range: 1,500-2,000
Armor Penetration (mm): 20 at 1,000 m/ 30 at 500 m

Other Ammunition Types: API, I-T

NOTES
BTR-80 is superior to BTR-60/70 with a larger chassis, high-angle-of- fire turret, and single more powerful diesel engine (vs gasoline). Options include the Kliver turret with a 30-mm gun, 7.62-mm coax MG, thermal sights, superior day sights, and (four) Kornet ATGM launchers.

Russian Armored Personnel Carrier BTR-80A



Weapons & Ammunition Types	Typical Combat Load
30-mm automatic gun HEI-T, Frag-HE-T AP-T, APDS-T, APFSDS-T	300
7.62-mm coax MG	2,000

SYSTEM
Alternative Designations: GAZ-59034
Date of Introduction: 1994
Proliferation: At least 3 countries
Description:
Crew: 2
Troop Capacity: 8
Combat Weight (mt): 14.6
Chassis Length Overall (m): 7.65
Height Overall (m): 2.80
Width Overall (m): 2.90
Ground Pressure (kg/cm^2): INA
Drive Formula: 8 x 8

Automotive Performance:
Engine Type: 260-hp Diesel
Cruising Range (km): 800
Speed (km/h):
Max Road: 90
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 10
Fording Depths (m): Amphibious

Radio: R-163-50U VHF, R-163-UP receiver, R-174 intercom

Protection:
Armor, Turret Front (mm): Can defeat 12.7-mm
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Mineclearing Equipment: No
Self-Entrenching Blade: N/A
Active Protective System: N/A
NBC Protection System: Collective
Smoke Equipment: 6 x 81-mm smoke grenade launchers

ARMAMENT
Main Armament:
Caliber, Type, Name: 30-mm automatic gun, 2A72
Rate of Fire (rd/min): 200-330 variable cyclic in bursts
Loader Type: Dual-belt feed
Ready/Stowed Rounds: 300/ 0
Elevation (°): -5 to +70
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm machinegun PKT
Mount Type: Coax
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,000 Night: 800+

Fire on Move: Yes
Rate of Fire (rd/min): 250 practical/650 cyclic, 2-10 round bursts

ATGM Launcher: N/A
Firing Ports: 4 right side, 3 left side

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: 2-plane
Rangefinder: INA
Infrared Searchlight: OU-5
Sights w/Magnification:
Gunner:
Day: 1P3-9, 1.2x/4x
Field of View (°): 49/14 (est)
Acquisition Range (m): 4,000
Night: TPN3-42 II/IR
Field of View (°): INA
Acquisition Range (m): 800
Commander Fire Main Gun: No

VARIANTS
BTR-80S: APC has the same turret with 14.5-mm vs 30-mm gun.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
30-mm APDS-T
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 2,000
Night: INA
Tactical AA Range: 2,500
Armor Penetration (mm): 25 (RHA) at 1,500 m

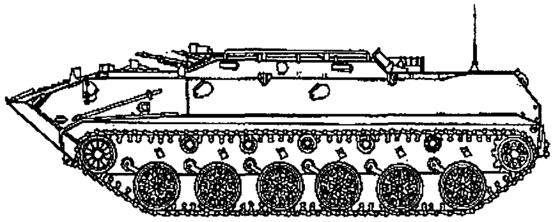
30-mm APFSDS-T, M929
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 2,000+
Night: INA
Tactical AA Range: 2,500
Armor penetration (mm): 55 (RHA) at 1,000 m/45 at 2,000 m

30-mm Frag-HE
Maximum Aimed Range (m): 4,000
Max Effective Range (m):
Day: 4,000
Night: INA
Tactical AA Range: 2,500
Armor Penetration (mm): INA

Other Ammunition Types: 30-mm AP-T, HEI-T

NOTES
The drop-in gun/turret package (Modular Weapon System) is offered for export, to upgrade a wide variety of vehicles to BTR-80A standard.
BTR-80A can mount K1-126 bullet-resistant tires.

Russian Airborne Armored Personnel Carrier BTR-D



Weapons & Ammunition Types	Typical Combat Load
2x 7.62-mm hatch MG Lt Ball, Ball-T, API, API-T, Incendiary	**2,000**

SYSTEM
Alternative Designations: BMD M1979
Date of Introduction: 1974
Proliferation: At least 1 country
Description:
Crew: 1
Troop Capacity: 12 passengers
Combat Weight (mt): 6.7
Chassis Length Overall (m): 5.88
Height Overall (m): 1.67
Width Overall (m): 2.63
Ground Pressure (kg/cm^2): 0.5

Automotive Performance:
Engine Type: 240-hp Diesel
Cruising Range (km): 500
Speed (km/h):
Max Road: 61
Max Off-Road: 35
Average Cross-Country: INA
Max Swim: 10
Fording Depth (m): Amphibious

Radio: R-123

Protection:
Armor, Turret Front (mm): "Antibullet" (7.62-mm)
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
Active Protective System: No
NBC Protection System: Yes
Smoke Equipment: 2x2 forward firing smoke grenade launchers
Vehicle engine exhaust smoke system (VEESS)

ARMAMENT
Main Armament:
Caliber, Type, Name: 7.62-mm (7.62 x 54R) machinegun, PKT
Rate of Fire (rd/min): 250 practical / 650 cyclic, in 2-10 round bursts
Loader Type: Belt-fed
Ready/Stowed Rounds: INA
Elevation (°): INA
Fire on Move: Yes

Auxiliary Weapon: N/A

ATGM Launcher: N/A
Firing Ports: 2 on each side, 1 in left rear door, permit two 5.45-mm RPK-74 light machineguns to be used.

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: N/A
Sights w/Magnification: Open, 1x
Commander Fire Main Gun: No

VARIANTS
1V118 Reostat/1V119 Spektr: Artillery command and observation posts for amphibious and airborne forces.

2S9: 120-mm self-propelled combination gun, with a turreted breech-loaded mortar/howitzer system.

BMD-KShM: Former Soviet regiment or division command and staff variant, with large Clothes-line antenna.

BREM-D: Armored repair and recovery variant.

BTR-RD/Robot: An ATGM variant (AT-4/-5) with 2 launchers, dismounted or mounted on pintles for vehicle launch.

BTR-ZD: Air defense variant with porteed or towed ZU-23 twin 23-mm air defense gun. Vehicle also carries manpad SAM launchers.

BTR-3D: Air defense variant with a rear-mounted ZU-23 gun.

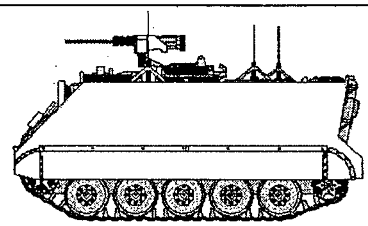
Sterkh (Malakit/Shmel): UAV transporter and launcher vehicle.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
7.62-mm API
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,000 m / 400-500 on the move
Night: INA
Tactical AA Range: INA
Armor Penetration (mm): 8 (RHA) at 500 m

Other Ammunition Types: 7.62-mm Light Ball, Ball-T, Heavy Ball, API, API-T, Incendiary

NOTES
BTR-D is a variant of the BMD-1, with an additional road wheel, with the turret removed, and with a raised hatch area. The vehicle can be parachute landed with airborne troops. The BTR-Ds in grenade launcher units will carry one AGS-17 30-mm AGL in the rear.
Options include the Kliver turret with a 30-mm gun, 7.62-mm coax MG, thermal sights, superior day sights, and (four) Kornet ATGM launchers.

US Armored Personnel Carrier M113A1



Weapons & Ammunition Types	Typical Combat Load
.50 cal cupola MG Slap, API, API-T, Ball, Ball-T, Incendiary, I-T	2,000

SYSTEM
Alternative Designations: INA
Date of Introduction: 1964
Proliferation: At least 46 countries
Description:
Crew: 2
Troop Capacity: 11 passengers
Combat Weight (mt): 11.20
Chassis Length Overall (m): 4.90
Height Overall (m): 1.80
Width Overall (m): 2.70
Ground Pressure (kg/cm^2): .55

Automotive Performance:
Engine Type: 215-hp Diesel
Cruising Range (km): 483
Speed (km/h):
Max Road: 64
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 5.8
Fording Depths (m): Amphibious

Radio: Various, including intercom

Protection:
Armor, Turret Front (mm): N/A—No turret
Applique Armor (mm): Yes—anti-mine armor on bottom
Explosive Reactive Armor (mm): Available
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: N/A
Smoke Equipment: N/A

ARMAMENT
Main Armament:
Caliber, Type, Name: .50 cal (12.7 x 99) heavy machinegun, M2HB
Rate of Fire (rd/min): 450-550 cyclic
Loader Type: Belt feed
Ready/Stowed Rounds: 250/1750
Elevation (°): -20/+60
Fire on Move: Yes

Auxiliary Weapon: N/A
ATGM Launcher: N/A
Firing Ports: None

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: N/A
Sights w/Magnification:
Gunner:
Day: Open ladder sight, 1x
Field of View (°): INA
Acquisition Range (m): 2,000
Night: N/A
Commander Fire Main Gun: No

VARIANTS
More than 100 variants have been produced in numerous countries, with 7.62-mm MGs, 40-mm automatic grenade launchers, 90-mm recoilless rifles, and turrets with 20-to-76-mm cannons. The following are US variants. Command variants are **M577** and **M114** (C&R). Self-propelled mortars are **M125** (81-mm) and **M106A2** (4.2-inch). The **M901/ ITV** ATGM launcher vehicle is a common variant. The **M163** SP air defense gun has a 20-mm Vulcan cannon; and **M730** is a Chapparral AD missile launcher. Other variants include ambulances, recovery and engineer vehicles.

M113A2: This multi-national variant features mobility improvements. One version is being developed with the Giat TS90 90-mm cannon.
M113A3: Changes for this multi-national variant include a new power train and increased armor protection.

AIFV: The multi-national IFV variant has M113A3 armor upgrades, a stabilized turret with 25-mm gun, and a 7.62-mm MG.

VCC-1: Italian M113 copies are supplemented by this variant

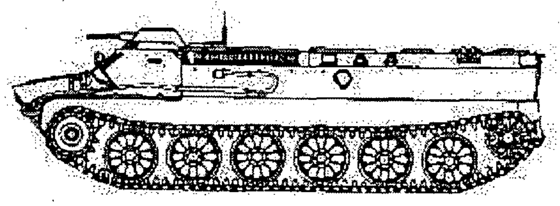
MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
.50 SLAP (sabot light armor penetrator)
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 2,000
Night: INA
Armor Penetration (mm): INA

.50 Cal Ball
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,000
Night: N/A
Armor Penetration (mm): INA

Other Ammunition Types: Ball-T, Incendiary, I-T, API, API-T

NOTES
The M113A1 is a variant of the gasoline-powered **M113**. Armors available include Rafael Enhanced Add-on Armor Kit (EAAK), Creusot-Marrel plate armor, and SNPE explosive reactive armor. Thermal and TV sights are also available.

Russian Light Armored Multi-purpose Vehicle MT-LB



Weapons & Ammunition Types	Typical Combat Load
7.62-mm Turret MG Lt Ball, Ball-T, API, API-T, Incendiary	2,000

SYSTEM
Alternative Designations: MT-LB-T
Date of Introduction: 1970, modernized in 1995
Proliferation: At least 9 countries
Description:
Crew: 2
Troop Capacity: 11 passengers
Combat Weight (mt): 11.9
Chassis Length Overall (m): 6.35
Height Overall (m): 1.87
Width Overall (m): 2.85
Ground Pressure (kg/cm^2): 0.46 standard track / 0.28 wide track

Automotive Performance:
Engine Type: 290-hp Diesel
Cruising Range (km): 500
Speed (km/h):
Max Road: 61.5/70 modernized
Max Off-Road: 30/45 modernized
Average Cross-Country: INA
Max Swim: 3-4
Fording Depth (m): Amphibious

Radio: R-123 or upgrade to -123M/-173

Protection:
Armor, Turret Front (mm): 7-14
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): INA
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: Yes
NBC Protection System: Collective
Smoke Equipment: N/A

ARMAMENT
Main Armament(s):
Caliber, Type, Name: 7.62-mm (7.62x 54R) machinegun, PKT
Rate of Fire (rd/min): 250 practical / 650 cyclic, 2-10 round bursts
Loader Type: Belt-feed
Ready/Stowed Rounds: INA
Elevation (°): -5/ +35
Fire on Move: Yes

Auxiliary Weapon: N/A
ATGM Launcher: N/A

Firing Ports: 1 on each side and 1 in each rear door.

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: N/A
Sights w/Magnification:
Gunner:
Day: PP-61AM, 2.6x
Field of View (°): 23
Acquisition Range (m): 1,500 (est)
Night: N/A
Commander Fire Main Gun: No

VARIANTS
MT-LB Upgrade: 1995 upgrade includes improved steering and a new engine.
2S1: 122-mm self-propelled howitzer.
9P149/Shturm-S: ATGM launcher vehicle with AT-6 autoloader.
MT-LB "blade": Dozer version with a blade attached to the vehicle.
MT-LBu: Expanded variant for artillery command and reconnaissance vehicles (ACRVs) and other uses.
MT-LBV: Arctic variant with .57m wide track for snow and improved flotation.
MTP-LB: Technical support vehicle.
MT-SON: Ground surveillance radar vehicle with Pork Trough/ SNAR-2 radar.
RKhM: Chemical reconnaissance vehicle.
SA-13: Regimental surface-to-air missile launcher vehicle.
SNAR-10: Ground surveillance radar vehicle with Big Fred radar.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
7.62-mm API, API-T
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,000/400-500 on the move
Night: INA
Tactical AA Range: INA
Armor Penetration (mm): 8 (RHA) at 500 m

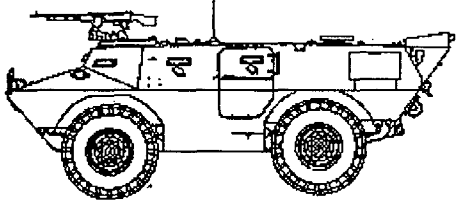
Other Ammunition Types: Light Ball, Ball-T, Heavy Ball, Incendiary

NOTES
Russian AG-17 30-mm automatic grenade launcher modification is available for use on MT-LB.

Russian KBP offers a drop-in one-man turret, called Kliver, with a stabilized 2A72 30-mm gun, a 4 Kornet ATGM launcher, thermal sights, and improved fire control system.

US Armored Personnel Carrier V-150

	Weapons & Ammunition Types	Typical Combat Load

	**7.62-mm cupola MG** Ball-T, Match API, API-T	**3,200**

SYSTEM
Alternative Designations: Commando
Date of Introduction: 1971
Proliferation: At least 20 countries
Description:
Crew: 3
Troop Capacity: 2
Combat Weight (mt): 9.89
Chassis Length Overall (m): 5.69
Height Overall (m): 1.98
Width Overall (m): 2.26
Ground Pressure (kg/cm^2): INA
Drive Formula: 4 x 4

Automotive Performance:
Engine Type: 202-hp Diesel
Cruising Range (km): 643
Speed (km/h):
Max Road: 89
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 5
Fording Depth (m): Amphibious

Radio: INA

Protection:
Armor, Turret Front (mm): Against 7.62-mm ball
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
Active Protective System: N/A
NBC Protection System: N/A
Smoke Equipment: Optional

ARMAMENT
Main Armament:
Caliber, Type, Name: 7.62-mm (7.62 x 51) MG, FN-MAG (example)
Rate of Fire (rd/min): 650-1000 cyclic
Loader Type: Belt feed, box magazines
Ready/Stowed Rounds: INA
Elevation (°): INA
Fire on Move: Yes

Auxiliary Weapon: N/A

ATGM Launcher: N/A
Firing Ports: None

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: N/A
Sights w/Magnification:
Gunner:
Day: Open ladder sight
Field of View (°): INA
Acquisition Range (m): 2,000
Night: N/A
Commander Fire Main Gun: No

VARIANTS
Armament options vary widely and include: a turret with 7.62-mm or 12.7-mm MG or turrets with 20-mm, 25-mm, 30-mm, 76-mm, or 90-mm gun. Another turret offers a 12.7-mm MG and 40-mm grenade launcher. An 81-mm self-propelled mortar launcher variant and a TOW ATGM launcher variant are available. Variants include a cargo carrier, police and security vehicles, an air defense variant with 20-mm Vulcan cannon.

V-100: This earlier 4x4 APC has a gasoline engine.

V-150S: This slightly larger 4x4 variant has improved drive train and the above variety of turret and gun options. A Commando command variant includes a raised compartment area with external-mount 7.62-mm MG. Taiwan has versions with an open-mount 12.7-mm MG and a 107-mm (4.2 inch) mortar.

V-200: Variant sold to Singapore with 20-mm turret, 90-mm turret, air defense variant with RBS-70 surface-to-air missile and a recovery variant. Another variant has a 120-mm mortar.

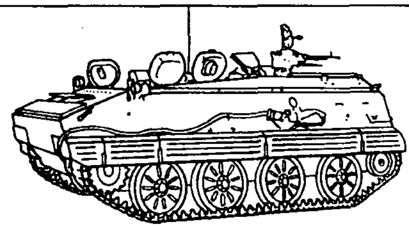
MAIN ARMAMENT AMMUNITION
Caliber, Type, Name: 7.62-mm Ball, NATO
Maximum Aimed Range (m): 2,000 (est)
Max Effective Range (m):
Day: 1,500
Night: INA
Tactical AA Range: INA
Armor Penetration (mm): INA

Other Ammunition Types: Ball-T, API, API-T, Match

NOTES
The baseline V-150 is equipped with a variety of pintle-mounted 7.62-mm machineguns. Many MGs are installed by user countries from their inventories. The Belgian FN-MAG general purpose MG is a widely used MG that represents a common capability.

Chinese Armored Personnel Carrier YW-531A

	Weapons & Ammunition Types	Typical Combat Load
	12.7-mm MG	**1,120**

	APDS, API, API-T, HE-T, HEI	500 620

SYSTEM
Alternative Designations: Type 63, North Korean M1967
Date of Introduction: Late 1960s
Proliferation: At least 9 countries
Description:
Crew: 4
Troop Capacity: 10 passengers
Combat Weight (mt): 12.60
Chassis Length Overall (m): 5.48
Height Overall (m): 2.85
Width Overall (m): 2.98
Ground Pressure (kg/cm^2): 0.44

Automotive Performance:
Engine Type: 320-hp Diesel
Cruising Range (km): 500
Speed (km/h):
- Max Road: 65
- Max Off-Road: 46
- Average Cross-Country: INA
- Max Swim: 6.0

Fording Depths (m): Ampphibious

Radio: INA

Protection:
Armor, Turret Front (mm): 14, front glacis
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
Active Protective System: NA
NBC Protection System: N/A
Smoke Equipment: N/A

ARMAMENT
Main Armament:
Caliber, Type, Name: 12.7-mm (12.7 x 108), heavy MG, Type 54
Rate of Fire (rd/min): 80-100 practical/600 air targets in bursts
Loader Type: Belt feed
Ready/Stowed Rounds: INA
Elevation (°): -4/+82
Fire on Move: Yes

Auxiliary Weapon: N/A
ATGM Launcher: N/A
Firing Ports: 1 on each side, and 1 in the rear

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: N/A
Sights w/Magnification:
Gunner:
- Day: Open ladder sight, NFI
  - Field of View (°): INA
  - Acquisition Range (m): 2,000
- Night: N/A

Commander Fire Main Gun: No

VARIANTS
M1973: North Korean variant, also known as VTT-323, has a turret with twin 14.5-mm guns. Some versions have Susong-Po (AT-3/ SAGGER variant) ATGM launcher and SA-7/16 manportable SAMs.

Type 54-1: Self-propelled 122-mm open-mount howitzer.
Type 70: Variant is a 130-mm (19-tube) multiple rocket launcher.
Type YW-304: Self-propelled 82-mm mortar.
Type YW-381: Self-propelled 120-mm mortar.
YW-750: Ambulance with a similar box compartment
YW-531C: This variant has a rectangular, three-sided open-topped shield around the gun, and better vision ports and ventilation. The C, D and E variants differ in intercom sets and firing ports.

Type YW-701: Command post variant with a box compartment over rear half of vehicle, and 5 radios. Armament is a 7.62-mm MG.

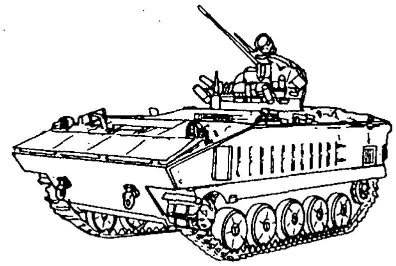
MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
12.7-mm, APDS (Tungsten Core), Type 54
- Maximum Aimed Range (m): 2,000
- Max Effective Range (m):
  - Day: 1,500 vehicles /1,600 aircraft
  - Night: INA
- Tactical AA Range: 1,600
- Armor Penetration (mm): INA

12.7-mm, API, Type 54
- Maximum Aimed Range (m): 2,000
- Max Effective Range (m):
  - Day: 1,500 unarmored ground / 800 armored
  - Night: INA
- Tactical AA Range: 1,000
- Armor Penetration (mm): 21 (RHA) at 500 m, 13 at 1,000 m

Other Ammunition Types: API-T, Russian Duplex, Russian Duplex-T, Incendiary-T, HE-T Type MDZ, HEI Type ZP

NOTES
Type 54 MG is a Chinese copy of former Soviet 12.7-mm DShKM.

French Infantry Fighting Vehicle AMX-10P



Weapons & Ammunition Types	Typical Combat Load
20-mm Cannon	**760**
APDS-T, API-T	(est) 260
HEI, HEI-T	500
7.62-mm coax MG	**2,000**
Tracer, AP, API, Incendiary	

SYSTEM
Alternative Designations: INA
Date of Introduction: 1973
Proliferation: At least 3 countries
Description:
Crew: 3
Troop Capacity: 8 passengers
Combat Weight (mt): 14.5
Length Overall (m): INA
Height Overall (m): 2.57
Width Overall (m): 2.78
Ground Pressure (kg/cm^2): 0.53

Automotive Performance:
Engine Type: 300-hp Diesel
Cruising Range (km): 600
Speed (km/h):
Max Road: 65
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 7
Fording Depths (m): Amphibious
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A

Radio: INA

Protection:
Armor, Turret Front (mm): 12.7-mm frontal
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): Available
Active Protective System: N/A
NBC Protection System: Yes
Smoke Equipment: 4 smoke grenade launchers

ARMAMENT
Caliber, Type, Name: 20-mm automatic cannon M693 F1
Rate of Fire (rd/min): 740
Loader Type: Dual belt feed
Ready/Stowed Rounds: INA
Elevation (°): -8/+50
Fire on Move: INA

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62 x 51) MG, AAT 52 NF1
Mount Type: Coax
Maximum Aimed Range (m): INA
Max Effective Range:
Day: INA
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): INA

ATGM Launcher: N/A
Firing Ports: None

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: INA
Rangefinder: INA
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: OB 40 Day/ night sight
Field of View (°): INA
Acquisition Range (m): INA
Night: OB 40 Day/ night sight
Field of View (°): INA
Acquisition Range (m): INA
Commander Fire Main Gun: No

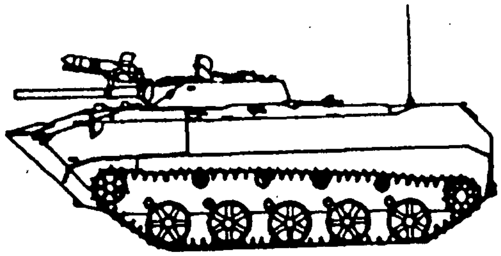
VARIANTS
AMX-10P: Variant with Milan or HOT ATGM launcher
AMX-10P/Milan: ATGM launcher vehicle, with two launchers
AMX/HOT: ATGM launcher vehicle (Toucan II turret, 4 launchers)
AMX-10 TM: Mortar carrier towing 120-mm RT-61 mortar
AMX-10 PAC 90: Fire support/AT variant with Giat 90-mm gun
AMX-10P Marine: Improved swim variant w/ 12.7/25/90-mm gun
AMX-10 PC: Command variant with varied command stations
AMX-10 RC: Wheeled (6 x 6) fire support vehicle with 90-mm gun
AMX-10 RAC: The same fire support chassis with 105-mm gun

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name: 20-mm (20x139) APDS-T
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 1,300
Night: INA
Tactical AA Range: INA
Armor Penetration (mm): INA

Other Ammunition Types: API, API-T, HEI, HEI-T

NOTES
A French SNPE explosive reactive armor (ERA) kit and others are available for use on theAMX-10P. However, during dismounted troop movement, ERA would be a hazard. Thus, passive armor is more likely; and ERA application is doubtful.

Russian Airborne Fighting Vehicle BMD-1



Weapons & Ammunition Types	Typical Combat Load
73-mm gun	**40**
HEAT	(est) 16
HE	24
AT-3/a/b/c/Imp ATGM	**4**
HEAT	3
HE	1
7.62-mm coax MG	**2,000**
2x 7.62-mm bow MG	**4,000**

SYSTEM
Alternative Designations:
Date of Introduction: 1969
Proliferation: At least 1 country
Description:
Crew: 2
Troop Capacity: 5 passengers (+1)
Combat Weight (mt): 13.3
Chassis Length Overall (m): 6.74
Height Overall (m): 2.15
Width Overall (m): 2.94
Ground Pressure (kg/cm^2): 0.57

Automotive Performance:
Engine Type: 240-hp Diesel
Cruising Range (km): 600
Speed (km/h):
Max Road: 65
Max Off-Road: 40-45
Average Cross-Country: INA
Max Swim: 7
Fording Depth (m): Amphibious

Radio: R-123M

Protection:
Armor, Turret Front (mm): 23 or Antibullet
Applique Armor (mm): See NOTES
Explosive Reactive Armor (mm): See NOTES
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: Collective
Smoke Equipment: VEESS

ARMAMENT
Main Armament:
Caliber, Type, Name: 73-mm smoothbore gun, 2A28/Grom
Rate of Fire(rd/min): 7-8
Loader Type: Autoloader
Ready/Stowed Rounds: 40 / 0
Elevation (°): -4/ +33
Fire on Move: Yes, but only 10 km/h or less (est)

Auxiliary Weapons:
Caliber, Type, Name: 7.62-mm (7.62x 54R) machinegun, PKT
Mount Type: Coax
Maximum Aimed Range (m): 1,300
Max Effective Range (m):
Day: 1,000/400-500 on the move
Night: 800
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, 2-10 round bursts
Caliber, Type, Name: 7.62-mm machinegun, PKT
Mount Type: Bow (ball-mounted)
Maximum Aimed Range (m): 1,000
Max Effective Range (m):
Day: 1,000/ 400-500 on the move
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, 2-10 round bursts

ATGM Launcher:
Name: 9P111
Launch Method: Rail-launched
Guidance: MCLOS
Command Link: Wire
Launcher Dismountable: Yes

Firing Ports: 1 on each side, 1 in left rear door

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: N/A
Rangefinder: Stadiametric
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: 1PN22M1, 8x
Field of View (°): 15
Acquisition Range (m):
Night: 1PN22M1, 6.7x
Field of View (°): 6
Acquisition Range (m): 800-1,000, based on light
Commander Fire Main Gun: No

VARIANTS
BMD-1K: Command IFV (FSU), with added R-126 and R-107.

BMD-1M: Variant with improved ventilation and road wheels.

BMD-1P: Widely fielded IFV with AT-4/5 replacing AT-3.

BMD-1PK: Company commander variant (FSU) of BMD-1P.

BMD-2: Widely fielded variant with a 30-mm automatic gun and with AT-3 ATGM launcher replaced by AT-4/5 ATGM launcher.

BTR-D: Stretched (6-road wheel) armored multipurpose transporter variant, with two 7.62-mm MGs but no turret. This chassis has been used for a variety of other airborne vehicles.

Russian Airborne Fighting Vehicle BMD-1 continued

MAIN ARMAMENT AMMUNITION	Antitank Guided Missiles:
Caliber, Type, Name: 73-mm HEAT-FS, PG-9 Maximum Aimed Range (m): 1,300 Max Effective Range (m): Day: 800, but 600 or less on the move Night: 800 Tactical AA Range: INA Armor Penetration (mm): 335 (RHA) 73-mm HEAT-FS, NFI Maximum Aimed Range (m): 1,300 Max Effective Range (m): Day: 1,000, but 600 or less on the move Night: 800-1,000 Tactical AA Range: INA Armor Penetration (mm): >400 (RHA) 73-mm HE, OG-9 Maximum Aimed Range (m): 1,300, 600 or less on the move Max Effective Range (m): Day: 1,300, but 600 or less on the move Night: 800-1,000 Tactical AA Range: INA Armor penetration (mm): INA 73-mm HE, OG-9M1 Maximum Aimed Range (m): 4,500 Max Effective Range (m): Day: 1,300, but 600 or less on the move Night: 800-1,000 Tactical AA Range: INA Armor Penetration (mm): INA **Other Ammunition Types:** OG-9M	Name: AT-3, -3A, -B Warhead Type: Tandem HEAT Armor Penetration (mm): 410 RHA Range (m): 3,000 Name: AT-3C Warhead Type: Tandem HEAT Armor Penetration (mm): 520 RHA Range (m): 3,000 Name: AT-3C Imp/ Polk (Slovenian) Warhead Type: Precursor with HEAT Armor Penetration (mm): 580 RHA Range (m): 3,000 Name: Malyutka-2 (Russian) Warhead Type: Tandem HEAT Armor Penetration (mm): 800 RHA Range (m): 3,000 Name: Malyutka HE (Russian) Warhead Type: Frag-HE Armor Penetration (mm): N/A Range (m): 3,000

NOTES

Vehicle can be parachute landed with airborne troops onboard. Height can be lowered.

Russian KBP offers a drop-in one-man turret, called Kliver, with a stabilized 2A72 30-mm gun, a 4-Kornet ATGM launcher, thermal sights, and improved fire control system. The Russian Volgorod Tractor Plant offers the B30 turret (a drop-in one-man turret with 2A42 30-mm gun, 7.62-mm coax MG, and a 9P135M launcher for AT-4/-5 ATGM). A Russian AG-17 30-mm automatic grenade launcher is available for BMD-1.

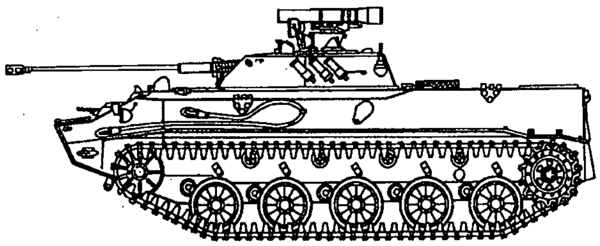
Other options are spall liners, air conditioning, and a more powerful engine. A French SNPE explosive reactive armor (ERA) kit and others are available for use on the BMD-1. However, during dismounted troop movement, ERA would be a hazard. Thus, passive armor is more likely; and ERA application is doubtful. For amphibious use, additional armor application is unlikely.

The Slovenian TS-M ATGM thermal night sight has a detection range of 4,500m and a recognition range of 2,000m.

The AT-3 HE-Blast ATGM is used for killing personnel and destroying bunkers and other fortifications.

The AT-3C Polk features a nose probe, an improved motor for increased velocity, lower smoke noise signature and a SACLOS launcher with improved sights.

Russian Airborne Fighting Vehicle BMD-3



Weapons & Ammunition Types	Typical Combat Load
30-mm automatic gun	**860**
HEI-T, Frag-HE	340/240
AP-T, APDS-T, APFSDS	160/120
ATGM launcher	**6**
7.62-mm coax MG	**2,000**
30-mm grenade launcher	**551**
5.45-mm light MG	**2,325**

SYSTEM
Alternative Designations: N/A
Date of Introduction: 1992
Proliferation: At least 1 country
Description:
Crew: 3
Troop Capacity: passengers: 4 (+3)
Combat Weight (mt): 12.9
Length Overall (m): 6.00
Height Overall (m): 2.25
Width Overall (m): 3.13
Ground Pressure (kg/cm^2): 0.32 (wide track) / .48 (standard track)

Automotive Performance:
Engine Type: 450-hp Diesel
Cruising Range (km): 500
Speed (km/h):
Max Road: 70
Max Off-Road: 45
Average Cross-Country: INA
Max Swim: 10
Fording Depth (m): Amphibious

Radio: R-173

Protection:
Armor, Turret Front (mm): "Antibullet" (7.62)
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): Available
Active Protective System: N/A
Mineclearing Equipment: KMT-8 plow or -10 roller
Self-Entrenching Blade: N/A
NBC Protection System: Collective
Smoke Equipment: Smoke grenade launchers, 3 x each side of turret
Vehicle engine exhaust smoke system (VEESS)

ARMAMENT
Main Armament:
Caliber, Type, Name: 30-mm automatic gun, 2A42
Rate of Fire (rd/min): 550 cyclic in bursts/ 200-300 practical
Loader Type: Dual-belt feed
Ready/Stowed Rounds: 500/ 360
Elevation (°): -5 to +74
Fire on Move: Yes

Auxiliary Weapons:
Caliber, Type, Name: 30-mm automatic grenade launcher, AG-17
Mount Type: Bow left side
Maximum Aimed Range (m): 1,700
Max Effective Range (m):
Day: 1,200
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 60-100 practical in short (≤ 5 rds) or long (6-10 rd) bursts.

Caliber, Type, Name: 7.62-mm (7.62 x 54R) machinegun, PKT
Mount Type: Turret coax
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,000
Night: 1,000
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, 2-10 round bursts

Caliber, Type, Name: 5.45-mm light machinegun, RPK-74
Mount Type: Bow right side
Maximum Aimed Range (m): 1,000
Max Effective Range (m):
Day: 800
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): 150 automatic/ 50 semiautomatic

ATGM Launcher:
Name: 9P135
Launch Method: Tube-launched
Guidance: SACLOS
Command Link: Wire
Launcher Dismountable: Yes

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: 2-plane
Rangefinder: Laser
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: BPK-2-42
Field of View (°): 8
Acquisition Range (m): INA
Night: BPK-2-42 II/IR
Field of View (°): INA
Acquisition Range (m): INA
Commander Fire Main Gun: No

VARIANTS None

Russian Airborne Fighting Vehicle BMD-3 continued

MAIN ARMAMENT AMMUNITION **Caliber, Type, Name:** 30-mm AP-T Maximum Aimed Range (m): 2,500 Max Effective Range (m): Day: 1,500 Night: INA Tactical AA Range: 4,000 Armor Penetration (mm): 18 (RHA) at 1,500m 30-mm APDS Maximum Aimed Range (m): 2,500 Max Effective Range (m): Day: 2,000 Night: INA Tactical AA Range: 4,000 Armor Penetration (mm): 25 (RHA) at 1,500m 30-mm APFSDS-T M929 Maximum Aimed Range (m): 2,500 Max Effective Range (m): Day: 2,000+ Night: INA Tactical AA Range: 4,000 Armor penetration (mm): 55 (RHA) at 1,000m, 45 at 2,000m	30-mm Frag-HE Maximum Aimed Range (m): 4,000 Max Effective Range (m): Day: 4,000 Night: INA Tactical AA Range: 4,000 Armor Penetration (mm): INA **Other Ammunition Types:** 30-mm HEI-T **Antitank Guided Missiles:** Name: AT-5B/Konkurs-M Warhead Type: Tandem shaped charge (HEAT) Armor Penetration (mm): 925 (RHA) Range (m): 4,000 Name: AT-5/Spandrel Warhead Type: Shaped charge (HEAT) Armor Penetration (mm): 650 (RHA) Range (m): 4,000

NOTES

BMD-3 has variable height control.

Automatic grenade launcher has 290 ready rounds and 261 in the rack. The ATGM launcher has 3 ready rounds (one on the launcher), and two stowed.

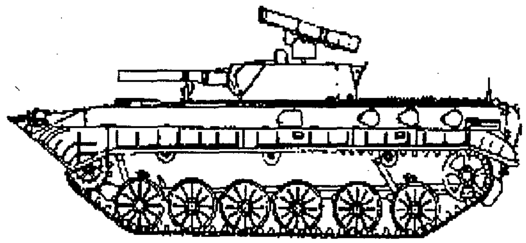
A French SNPE explosive reactive armor (ERA) kit and others are available for use on the BMD-3. However, during dismounted troop movement, ERA would be a hazard. Thus, passive armor is more likely and ERA application is doubtful. For amphibious use, additional armor application is unlikely. Other options are spall liners, air conditioning, and a more powerful engine.

The Russian SANOET-1 thermal gunner's sight is available. Thermal sights are available for the ATGM launcher. The Russian Trakt/1PN65 thermal imaging ATGM night sight is optional. Acquisition range is 2,500 m (NFI). For the ATGM launcher in dismount configuration, the Russian Mulat/1PN86 lightweight thermal ATGM night sight has 3,600 m detection range and 2,000 m identification range.

French-German Flame-V adapter kit permits the BMD-3 to launch Milan, Milan-2 and Milan-3 ATGMs.

Russian KBP offers a drop-in one-man turret, called Kliver, with a stabilized 2A72 30-mm gun, a 4 Kornet ATGM launcher, thermal sights, a coaxial 7.62-mm MG and improved fire control system.

Russian Infantry Fighting Vehicle BMP-1P



Weapons & Ammunition Types	Typical Combat Load
73-mm gun	**40**
HEAT-FS	(est) 16
HE	24
ATGM	**4**
AT-4/-4B/-5/-5B	
7.62-mm coax MG	**2,000**

SYSTEM
Alternative Designations: BWP-1 (Poland), see NOTES
Date of Introduction: 1974
Proliferation: At least 7 countries
Description:
Crew: 3
Troop Capacity: 6 passengers
Combat Weight (mt): 13.3
Chassis Length Overall (m): 6.74
Height Overall (m): 2.15
Width Overall (m): 2.94
Ground Pressure (kg/cm^2): 0.57

Automotive Performance:
Engine Type: 300-hp Diesel
Cruising Range (km): 600
Speed (km/h):
Max Road: 65
Max Off-Road: 40-45
Average Cross-Country: INA
Max Swim: 7
Fording Depth (m): Amphibious

Radio: R-123, or R-173

Protection:
Armor, Turret Front (mm): 19-23
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): Available
Active Protective System: N/A
Mineclearing Equipment: KMT-8 plow available
Self-Entrenching Blade: N/A
NBC Protection System: Collective
Smoke Equipment: Six 81-mm smoke grenade launchers, VEESS

ARMAMENT
Main Armament:
Caliber, Type, Name: 73-mm smoothbore gun 2A28/Grom
Rate of Fire (rd/min): 7-8
Loader Type: Autoloader
Ready/Stowed Rounds: 40 / 0
Elevation (°): -4/+33
Fire on Move: Yes, but only 10 km/h or less (est)

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) machinegun, PKT
Mount Type: Coax
Maximum Aimed Range (m): 1,300
Max Effective Range (m):
Day: 1,000 / 400-500 on the move
Night: 800
Fire on Move: No
Rate of Fire (rd/min): 250 practical / 650 cyclic, 2-10 round bursts

ATGM Launcher:
Name: 9P135M2
Launch Method: Tube-launched
Guidance: SACLOS
Command Link: Wire
Launcher Dismountable: Yes

Firing Ports: 4 on each side, 1 in left rear door

FIRE CONTROL
FCS Name: 1PN22M1
Main Gun Stabilization: N/A
Rangefinder: Stadiametric
Infrared Searchlight: OU-3GK
Sights w/Magnification:
Gunner:
Day: 1PN22M1, 8 x
Field of View (°): 15
Acquisition Range (m): 1,300
Night: 1PN22M1, 6.7x
Field of View (°): 6
Acquisition Range (m): 800-1,000 based on light
Commander Fire Main Gun: No

VARIANTS
BMP-1PG: This recently offered Russian upgrade is similar to BMP-1P with an added AG-17 30-mm automatic grenade launcher and other options, including thermal sights.

BMP-1PK: Command variant, with addition of R-126 and R-107 transceiver. A small telescoping antenna is mounted on right rear. Firing ports and telescopes on right side are blocked off.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
73-mm HEAT-FS, PG-9
Maximum Aimed Range (m): 1,300
Max Effective Range (m):
Day: 800, but 600 or less on the move in 2-4 round bursts
Night: 800-1,000
Tactical AA Range: INA
Armor Penetration (mm): 335 (RHA)

73-mm HEAT-FS, NFI
Maximum Aimed Range (m): 1,300
Max Effective Range (m):
Day: 1,000/ 600 or less on the move
Night: 800-1,000
Tactical AA Range: INA
Armor Penetration (mm): >400 (RHA)

Russian Infantry Fighting Vehicle BMP-1P continued

73-mm HE, OG-9M1 Maximum Aimed Range (m): 4,500 Max Effective Range (m): Day: 1,300/ 600-1,000 on the move Night: 800-1,000 Tactical AA Range: INA Armor penetration (mm): INA **Other Ammunition Types:** OG-9, OG-9M **Antitank Guided Missiles:** Name: AT-5/SPANDREL Warhead Type: Shaped charge (HEAT) Armor Penetration (mm): 650 (RHA) Range (m): 4,000	Name: AT-5B/Konkurs-M Warhead Type: Tandem shaped charge (HEAT) Armor Penetration (mm): 925 (RHA) Range (m): 4,000 Name: AT-4/SPIGOT Warhead Type: Shaped charge (HEAT) Armor Penetration (mm): 480 (RHA) Range (m): 2,000 Name: AT-4B/Factoria Warhead Type: Tandem Shaped charge (HEAT) Armor Penetration (mm): 550 (RHA) Range (m): 2,500

NOTES

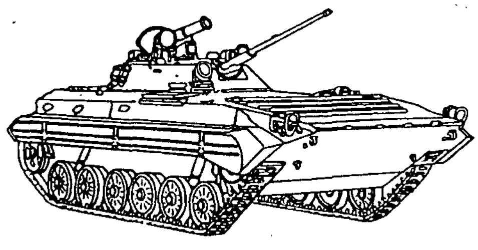
The prototype IFV, known as BMP, was not fielded. Initial BMP production variant, BMP-A, was halted with insignificant numbers. The baseline production IFV, BMP-1, has an AT-3/SAGGER antitank guided missile. The BMP-1P upgrade is widely fielded, with an AT-4/-5 ATGM launcher replacing the AT-3 launcher. The vehicle also added smoke grenade launchers. This variant should generally be portrayed where OPFOR calls for the BMP-1. For applications where a robust and modernized OPFOR is expected, use AT-5B ATGM. The AT-4/-4B ATGMs are less likely to be employed on this vehicle.

Other options are spall liners, air conditioning, and a more powerful engine. A French SNPE explosive reactive armor (ERA) kit and others are available for use on the BMD-1. However, during dismounted troop movement, ERA would be a hazard. Thus, passive armor is more likely; and ERA application is doubtful. Additional armor application may jeopardize amphibious capability.

Russian AG-17 30-mm automatic grenade launcher modification is available for use on BMP-1P. Russian KBP offers a drop-in one man turret, called Kliver, with a stabilized 2A72 30-mm gun, a 4 Kornet ATGM launcher, thermal sights, and improved fire control system.

The Russian Alis thermal gunner's sight is available. The Slovenian TS-F ATGM thermal night sight has a detection range of 4,500 m and a recognition range of 2,000 m.

Russian Infantry Fighting Vehicle BMP-2



Weapons & Ammunition Types	Typical Combat Load
30-mm automatic gun	**500**
HEI-T, Frag-HE	340
AP-T, APDS-T, APFSDS-T	160
ATGM	**5**
AT-5/-5B/-4/-4B	
7.62-mm coax MG	**2,000**

SYSTEM
Alternative Designations: Yozh (Russia), Sarath (India)
Date of Introduction: 1980
Proliferation: At least 20 countries
Description:
Crew: 3
Troop Capacity: 7 passengers
Combat Weight (mt): 14.3
Chassis Length Overall (m): 6.72
Height Overall (m): 2.45
Width Overall (m): 3.15
Ground Pressure (kg/cm^2): 0.63

Automotive Performance:
Engine Type: 300-hp Diesel
Cruising Range (km): 600
Speed (km/h):
Max Road: 65
Max Off-Road: 45
Average Cross-Country: 35
Max Swim: 7
Fording Depth (m): Amphibious

Radio: R-123M transceiver or R-173

Protection:
Armor, Turret Front (mm): 23-33
Applique Armor (mm): On BMP-2D
Explosive Reactive Armor (mm): Available, see NOTES
Active Protective System: N/A
Mineclearing Equipment: KMT-8 mine plow available
Self-Entrenching Blade: N/A
NBC Protection System: Collective
Smoke Equipment: 6 smoke grenade launchers, VEESS

ARMAMENT
Main Armament:
Caliber, Type, Name: 30-mm automatic gun, 2A42
Rate of Fire (rd/min): 550 cyclic in bursts/ 200-300 practical
Loader Type: Dual-belt feed
Ready/Stowed Rounds: 500/0
Elevation (°): -5 to +74
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) machinegun, PKT
Mount Type: Turret coax
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,000
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical/650 cyclic, 2-10 round bursts

ATGM Launcher:
Name: 9P135M1/M3
Launch Method: Tube-launched
Guidance: SACLOS
Command Link: Wire
Launcher Dismountable: Yes

Firing Port: 4 on left side, 3 on right side
1 in left rear door

FIRE CONTROL
FCS Name: BPK-1-42 or BPK-2-42
Main Gun Stabilization: 2-plane
Rangefinder: Laser
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: BPK-1-42 or BPK-2-42
Field of View (°): 8
Acquisition Range (m): 2,500-4,000 (est)
Night: BPK-1-42 or BPK-2-42 II/IR
Field of View (°): INA
Acquisition Range (m): INA
Commander Fire Main Gun: No

VARIANTS
BMP-2D: Variant with add-on plate armor, but which cannot swim

BMP-2E: Variant with 6-mm steel plates added and track skirts

BMP-2K: Command variant with additional radio

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
30-mm AP-T
Maximum Aimed Range (m): 2,500
Max Effective Range (m):
Day: 1,500
Night: INA
Tactical AA Range: 4,000
Armor Penetration (mm): 18 (RHA, 60°) at 1,500 m

Russian Infantry Fighting Vehicle BMP-2 continued

30-mm APDS Maximum Aimed Range (m): 2,500 Max Effective Range (m): Day: 2,000 Night: INA Tactical AA Range: 4,000 Armor Penetration (mm): 25 (RHA) at 1,500m 30-mm APFSDS-T M929 Maximum Aimed Range (m): 2,500 Max Effective Range (m): Day: 2,000+ Night: INA Tactical AA Range: 4,000 Armor penetration (mm): 55 (RHA) at 1,000m/45 at 2,000m 30-mm Frag-HE Maximum Aimed Range (m): 4,000/ 2,500 point target Max Effective Range (m): Day: 4,000 Night: INA Tactical AA Range: 4,000 Armor Penetration (mm): INA **Other Ammunition Types:** 30-mm HEI-T	**Antitank Guided Missiles:** Name: AT-5/SPANDREL Warhead Type: Shaped charge (HEAT) Armor Penetration (mm): 650 (RHA) Range (m): 4,000 Name: AT-5B/Konkurs-M Warhead Type: Tandem shaped charge (HEAT) Armor Penetration (mm): 925 (RHA) Range (m): 4,000 Name: AT-4/SPIGOT Warhead Type: Shaped charge (HEAT) Armor Penetration (mm): 480 (RHA) Range (m): 2,000 Name: AT-4B/Factoria Warhead Type: Tandem shaped charge (HEAT) Armor Penetration (mm): 550 (RHA) Range (m): 2,500

NOTES

A French SNPE explosive reactive armor (ERA) kit and others are available for use on the BMP-2. However, during dismounted troop movement, ERA would be a hazard. Thus, passive armor is more likely and ERA application is doubtful. For amphibious use, additional armor application is unlikely. Other options are spall liners, air conditioning, and a more powerful engine.

Russian AG-17 30-mm automatic grenade launcher modification is offered for BMP-2.

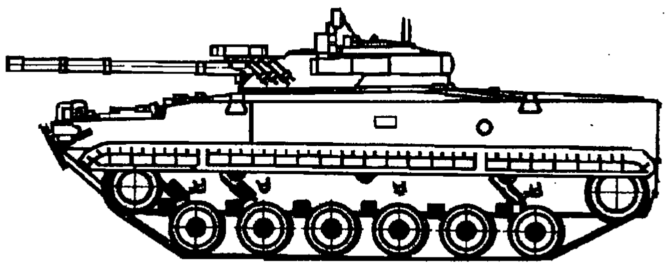
Russian KBP offers a drop-in one-man turret, called Kliver, with a stabilized 2A72 30-mm gun, a 4 Kornet ATGM launcher, thermal sights, a coaxial 7.62-mm MG and improved fire control system.

ATGM load consists of one ready on the launcher and four stowed. They are readily accessible, but require hand loading from an open hatch. The AT-5 and AT-5B are more likely than AT-4 and -4B.

French-German Flame-V adaptor kit permits the BMP-2 system to launch Milan, Milan-2, and Milan-3 ATGMs.

Thermal sights are available. The Russian SANOET-1 thermal gunner's sight is available. The Russian Trakt/1PN65 thermal imaging (TI) ATGM night sight is optional. Acquisition range is 2,500 m (NFI). For the launcher in dismount configuration, the Slovenian TS-F ATGM night sight is available and has a detection range of 4,500 m and recognition range of 2,000 m. The Russian Mulat/1PN86 lightweight TI ATGM thermal sight has 3,600 m detection range and 2,000 m identification range.

Russian Infantry Fighting Vehicle BMP-3



Weapons & Ammunition Types	Typical Combat Load
100-mm rifled gun	**40**
Frag-HE	40
AT-10/Imp ATGM	8
30-mm automatic gun	**500**
HEI-T, Frag-HE	340
AP-T, APDS-T or APFSDS-T	160
7.62-mm coax MG	**2,000**
2 x 7.62-mm bow MG	**4,000**

SYSTEM
Alternative Designations: Soviet ICV M1990/1
Date of Introduction: 1990
Proliferation: At least 7 countries
Description:
Crew: 3
Troop Capacity: 7 passengers
Combat Weight (mt): 18.70
Chassis Length Overall (m): 6.73
Height Overall (m): 2.45
Width Overall (m): 3.15
Ground Pressure (kg/cm²): 0.62

Automotive Performance:
Engine Type: 500-hp Diesel
Cruising Range (km): 600
Speed (km/h):
Max Road: 70
Max Off-Road: 45
Average Cross-Country: 35
Max Swim: 10
Fording Depth (m): Amphibious

Radio: R-173, R-173P

Protection:
Armor, Turret Front (mm): 30-35 front glacis
Applique Armor (mm): Yes on turret
Explosive Reactive Armor (mm): Available, see NOTES
Active Protective System: N/A
Mineclearing Equipment: KMT-8 plow available
Self-Entrenching Blade: Yes
NBC Protection System: Collective
Smoke Equipment: 6 smoke grenade launchers, VEESS

ARMAMENT
Main Armaments:
Caliber, Type, Name: 100-mm rifled gun 2A70
Rate of Fire (rd/min): 8-10
Loader Type: Autoloader gun rounds; manual for gun and ATGMs
Ready/Stowed Rounds: 22/18 for rounds, 3/5 for ATGMs
Elevation (°): -5 to +60
Fire on Move: Yes

Caliber, Type, Name: 30-mm automatic gun 2A72
Rate of Fire: 350 rd/min (cyclic) in bursts
Loader Type: Dual-belt feed
Ready/Stowed Rounds: 500/ 0
Elevation (°): -5 to +60
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) machinegun, PKT
Mount Type: Turret coax
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,000
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, in 2-10 round bursts

Caliber, Type, Name: 7.62-mm (7.62x 54R) machinegun, PKT
Mount Type: Bow left and right
Maximum Aimed Range (m): 1,000
Max Effective Range (m):
Day: 1,000/400-500 on the move
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, in 2-10 round bursts

ATGM Launcher:
Name: 2A70 100-mm gun
Launch Method: Gun-launched
Guidance: SACLOS, laser-beam rider
Command Link: Encoded infrared laser-beam
Launcher Dismountable: No

Firing Ports: 2 on each side, 1 in left rear door

FIRE CONTROL
FCS Name: 1K13-2
Main Gun Stabilization: 2-plane
Rangefinder: Laser
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: 1K13-2, 8x ; 1P3-10 antiaircraft, 2.6x; PPD-1 stand-by
Field of View (°): INA
Acquisition Range (m): 5,200
Night: 1K13-2 II night channel, 5.5x
Field of View (°): INA
Acquisition Range (m): INA
Commander Fire Main Gun: Yes

Russian Infantry Fighting Vehicle BMP-3 continued

VARIANTS
BMP-3F: Amphibious Armored Combat Vehicle developed for Naval Infantry.

BMP-3 M1995: ATGM launcher vehicle, with Kornet (AT-14) launcher and autoloader, and thermal sights.

9P157: ATGM launcher vehicle, with Krizantema (AT-15) ATGM autoloader, MMW and thermal fire control system.

BMP-3K: Command variant; with electronic round fuze system for 100-mm gun. Bow MGs are removed. Added radios are R-159, R-143 and R-174.

BREhM-L: Armored recovery vehicle (ARV).

BRM-3K: Combat recon vehicle with radar and 30-mm gun.

BMP-3: UAE upgrade improvements including Namut Thermal Night sight.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
100-mm HE 3UOF17
  Maximum Aimed Range (m): 5,000
  Max Effective Range (m):
    Day: 4,000
    Night: INA
  Tactical AA Range: 4,000
  Armor Penetration (mm): 25 (RHA)

Caliber, Type, Name: 100-mm HE-Shapnel (HEF/MOD.96)
    Focused-fragmentation, electronically-fuzed
  Maximum Aimed Range (m): 5,200
  Max Effective Range (m):
    Day: 5,200
    Night: INA
  Tactical AA Range: 4,000
  Armor Penetration (mm): INA

30-mm APFSDS-T M929
  Maximum Aimed Range (m): 2,500
  Max Effective Range (m):
    Day: 2,000+
    Night: INA
  Tactical AA Range: 4,000
  Armor penetration (mm): 55 (RHA) at 1,000 m, 45 at 2,000 m

30-mm Frag-HE
  Maximum Aimed Range (m): 4,000
  Max Effective Range (m):
    Day: 4,000
    Night: INA
  Tactical AA Range: 4,000
  Armor Penetration (mm): INA

30-mm AP-T
  Maximum Aimed Range (m): 2,500
  Max Effective Range (m):
    Day: 1,500
    Night: INA
  Tactical AA Range: 4,000
  Armor Penetration (mm): 18 (RHA, 60°) at 1,500 m

30-mm APDS
  Maximum Aimed Range (m): 2,500
  Max Effective Range (m):
    Day: 2,000
    Night: INA
  Tactical AA Range: 4,000
  Armor Penetration (mm): 25 (RHA) at 1,500 m

Other Ammunition Types: 100-mm HE-I, 30-mm HEI-T

Antitank Guided Missiles
Name: AT-10/Basnya
Warhead Type: Shaped charge
Command Link: Encoded laser-beam
Warhead Type: Shaped charge (HEAT)
Armor Penetration (mm): 650 (RHA)
Range (m): 4,000

Name: AT-10 Improved
Warhead Type: Tandem shaped charge
Armor Penetration (mm): 700 (RHA) behind ERA
Range (m): 4,000
Launcher Dismountable: No

NOTES
A French SNPE ERA kit and others are available for use on the BMP-3. However, during dismounted troop movement ERA would be a hazard. Thus, passive armor is more likely and ERA application is doubtful. Other options are spall liners and air conditioning.

Russian AG-17 30-mm automatic grenade launcher modification is available for use on BMP-3.

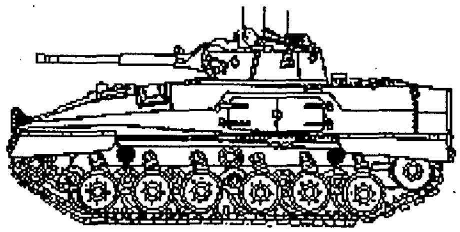
Russian KBP offers a drop-in one-man turret called Kliver, with a stabilized 2A72 30-mm gun, a 4 Kornet ATGM launcher, thermal sights, and improved fire control system.

The Namut thermal gunner's sight is available for use on BMP-3. This uses the French Athos thermal camera. Namut sight has 3x and 10x channels. Night acquisition range: 2,600 m (NFI)

Stowed rounds and ATGMs can be passed from the passenger compartment to the gunner for hand loading. This includes ATGMs.

The "HEF" (or "HE-Shrapnel") round can be employed in indirect fire mode with air burst to 7,000 m.

British Infantry Fighting Vehicle Warrior



Weapons & Ammunition Types	Typical Combat Load
30-mm auto gun	**228**
HEI-T	
APDS-T, APSE-T	
7.62-mm coax MG	**2,200**
Ball, Ball-T	

SYSTEM
Alternative Designations: FV 511, MCV-80
Date of Introduction: 1988
Proliferation: At least two countries
Description:
Crew: 3
Troop Capacity: 7 passengers
Combat Weight (mt): 24.00
Chassis Length Overall (m): 6.34
Height Overall (m): 2.79
Width Overall (m): 3.03
Ground Pressure (kg/cm^2): 0.65

Automotive Performance:
Engine Type: 550-hp Diesel
Cruising Range (km): 660
Speed (km/h):
Max Road: 75
Max Off-Road: 60
Cross-Country: 48
Max Swim: N/A
Fording Depth (m): 1.3 Unprepared

Radio: INA

Protection:
Armor, Turret Front (mm): Against 14.5-mm gun
Applique Armor (mm): Available (see VARIANTS)
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: Yes
Smoke Equipment: Smoke grenade launchers (4 each side of turret)

ARMAMENT
Main Armament:
Caliber, Type, Name: 30-mm automatic cannon, RARDEN L21A1
Rate of Fire (rd/min): 80-90 cyclic
Loader Type: Feed tray, clip-fed (3-round clips)
Ready/Stowed Rounds: 228/0
Elevation (°): -10/+45
Fire on Move: INA

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm chain gun, L94A1
Mount Type: Turret coax
Maximum Aimed Range (m): INA
Max Effective Range: INA
Fire on Move: Yes
Rate of Fire (rd/min): 520-570

ATGM Launcher: N/A
Firing Ports: None

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: N/A
Rangefinder: INA
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: INA
Field of View (°): INA
Acquisition Range (m): INA
Night: SPAV L2A1 II sight
Field of View (°): INA
Acquisition Range (m): INA
Commander Fire Main Gun: No

VARIANTS
Command variant is outfitted with radios, mapboards, other staff support equipment, and Vickers Defence Turret.

Desert Warrior: Variant with the 2-man turret from LAV-25, with a US M242 Bushmaster 25-mm automatic cannon, coaxial MG and 1-2 ATGM launchers. Other modifications are additional passive armor and three periscopes for improved vision. Sold to Kuwait.

Desert Storm Variant: Changes included passive armor added to hull sides and a pintle mount for a Milan-2 ATGM launcher.

Mechanized Artillery Observation Vehicle (MAOV): It resembles an IFV, but is fitted with a dummy cannon, improved artillery reconnaissance and automation systems, and land navigation. Options include an Osprey 8-power optical and thermal sight with Nd:YAG laser designator for the observer.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
30-mm APDS-T, L14
Maximum Aimed Range (m): 4,000
Max Effective Range (m):
Day: 1,100
Night: INA
Armor Penetration (mm): INA

Other Ammunition Types: 30-mm APSE-T (AP Secondary Effects-T L5, HEI-T L13

NOTES
Variants available but not in production include engineer, recovery,mortar vehicles, armored fighting vehicles with 90-mm and 105-mm guns, an APC with 7.62-mm chain gun, ATGM launcher vehicles for Milan, HOT and Trigat, and a low-profile chassis for a reduced signature IFV.

Chapter 3
Reconnaissance Vehicles

The modern battlefield is becoming increasingly mobile and lethal. The challenge for reconnaissance systems is to acquire the enemy, transmit intelligence, and survive for the next mission. Therefore, ground forces use specialized reconnaissance vehicles. Most will employ a mix of systems, including tanks and infantry fighting vehicles, dismounted reconnaissance patrols, aerial reconnaissance, and reconnaissance vehicles. The spectrum of reconnaissance vehicles currently ranges from older systems ill-suited for modern requirements, to survivable, mobile, and lethal systems, equipped with complex sensor arrays and communications suites.

A number of forces fielded *combat reconnaissance vehicles* (CRVs) designed for operations at or beyond the FLOT, not to initiate combat but to survive if engaged. They may operate in combat reconnaissance patrols with heavily armed vehicles such as tanks and IFVs. Many offer sensors no better than those on other armored vehicles, and use optics for a variety of combat support missions, such as fire support. Examples of these are the British Saladin Armored Car and the Austrian Pandur armored reconnaissance Fire Support Vehicle. Main guns on these vehicles can range up to 105 mm (South African Rooikat). A growing trend is for CRVs with added sensors (such as the Russian BRM-3K). It is a versatile vehicle configured for maneuver reconnaissance with thermal sights and a 30-mm gun, but is also useful for setting up a stationary surveillance position with its Tall Mike radar. As a command (-K type) vehicle, it employs a mix of radios to transmit intelligence across several nets in a combined arms force.

A recent trend is the fielding of vehicles with sophisticated multi-sensor arrays specially designed to operate behind or near the FLOT and provide continuous data to combined arms forces. An example is the Czech Snezka, which will be featured in an update. Vehicles designed to support specific branches are included with those branches (such as PRP-3/4 for artillery).

A class of vehicles widely proliferated for light patrol duties is the armored scout car. With wheels rather than tracks, light armor, and guns generally of 7.62 - 20 mm, they offer low cost but are vulnerable to a wide variety of weapons. Examples include the British Ferret and Russian BRDM-2. A recent category of vehicle which US Army forces will encounter is lightly armored vehicles on truck or jeep-type chassis with very light armor for security, and patrol. Some are unarmed; whereas others employ sophisticated weapons stations and lethal firepower (up to 30-mm guns). Smaller 4x4 scout vehicles (such as French VBL) and ultra-light fast-attack vehicles have also been built for light patrol and rapid reconnaissance missions.

This chapter provides a representative sampling of reconnaissance vehicles in use today. The selection is not comprehensive, rather reflects systems currently available to the OPFOR.

Questions and comments on data listed in this chapter should be addressed to:

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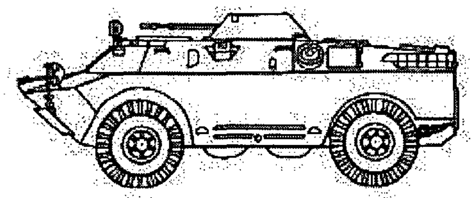
Russian Armored Scout Car BRDM-2



Weapons & Ammunition Types	Typical Combat Load
14.5-mm machinegun	**500**
API, API-T, I-T	160
HE-T	340
7.62-mm coax MG	**2,000**
Heavy ball, I-T, Light ball, Ball-T, API-T	

SYSTEM
Alternative Designations: GAZ 41-08
Date of Introduction: 1963
Proliferation: At least 45 countries
Description:
Crew: 4
Troop Capacity: 0 (for this configuration)
Combat Weight (mt): 7.0
Chassis Length Overall (m): 5.75
Height Overall (m): 2.31
Width Overall (m): 2.75
Ground Pressure (kg/cm^2): INA
Drive Formula: 4 x 4 (+ 4 auxiliary wheels)

Automotive Performance:
Engine Type: 140-hp Gasoline
Cruising Range (km): 750
Speed (km/h): Max Road: 95
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 10
Fording Depths (m): Amphibious

Radio: R-123

Protection:
Armor, Turret Front (mm): 10
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
Active Protective System: N/A
NBC Protection System: Collective
Smoke Equipment: N/A

ARMAMENT
Main Armament:
Caliber, Type, Name: 14.5-mm Machinegun KPVT
Rate of Fire (rd/min): 150 practical/600 cyclic
Loader Type: Belt feed
Ready/Stowed Rounds: 500/0
Elevation (°): -5 / +30
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) Machinegun PKT
Mount Type: Coax
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,000m / 400-500 on the move
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic in 2-10 round bursts

ATGM Launcher: N/A
Firing Ports: INA

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: INA
Sights w/Magnification:
Gunner:
Day: PP-61AM
Field of View (°): 23
Acquisition Range (m): 2,000
Night: N/A
Field of View (°): INA
Acquisition Range (m): INA
Commander Fire Main Gun: No

VARIANTS
Polish modernized variant has been offered in five versions, with upgrades such as: 12.7-mm MG, improved PKT MG mount, roof-mounted AT-4 ATGM, surveillance radar, improved fire control, etc.

Antitank guided missile launcher vehicles:
9P122 variant with 6-rail AT-3 ATGM launcher
9P124 variant with 4-rail AT-2 MCLOS ATGM launcher
9P133 variant with 6-rail AT-3C SACLOS ATGM launcher
9P137 variant with 5-rail AT-5 ATGM launcher
9P148 variant with 5-rail AT-4 or AT-5 ATGM launcher.

BRDM-2Rkh: NBC reconnaissance vehicle

BRDM-2U: Command variant without a turret

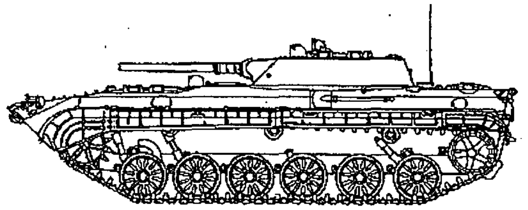
SA-9: SAM system transporter-erector-launcher vehicle

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
14.5-mm API-T
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 2,000
Night: INA
Tactical AA Range: 1,400
Armor Penetration (mm): 20 at 1,000 m/30 at 500 m

Other Ammunition Types: 14.5-mm API, I-T, HE-T Type MDZ

NOTES
Some BRDMs may include an AT-4 launcher and ATGMs for dismounted self-defense.

Russian Armored Reconnaissance Command Vehicle BRM-1K



Weapons & Ammunition Types	Typical Combat Load
73-mm gun	**20**
HEAT	(est) 10
HE	10
7.62-mm coax MG	**2,000**

SYSTEM
Alternative Designations: BMP M1976/2
Date of Introduction: 1976
Proliferation: At least 3 countries
Description:
Crew: 4 (with addition of a navigator)
Troop Capacity: 6 passengers
Combat Weight (mt): 13.3
Chassis Length Overall (m): 6.74
Height Overall (m): 2.15
Width Overall (m): 2.94
Ground Pressure (kg/cm^2): 0.57

Automotive Performance:
Engine Type: 300-hp diesel
Cruising Range (km): 600
Speed (km/h):
Max Road: 65
Max Off-Road: 40-45
Average Cross-Country: INA
Max Swim: 7
Fording Depth (m): Amphibious
Radio: R-173, R-130, 2x R-148 manportable, R-014D telegraph

Protection:
Armor, Turret Front (mm): 19-23
Applique Armor (mm): Available
Explosive Reactive Armor (mm): Available
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: Yes
Smoke Equipment: VEESS

ARMAMENT
Main Armament:
Caliber, Type, Name: 73-mm smoothbore gun, 2A28/Grom
Rate of Fire (rd/min): 7-8
Loader Type: Autoloader
Ready/Stowed Rounds: 20 / 0
Elevation (°) : -4/+33
Fire on Move: Yes, but only 10 km/h or less (est)

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) machinegun PKT
Mount Type: Coaxial
Maximum Aimed Range (m): 1,300
Max Effective Range (m):
Day: 1,000 / 400-500 on the move
Night: 800
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, in 2-10 round bursts
Firing Ports: 1 on each side, 1 in left rear door

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: No
Rangefinder: Laser
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: 1PN22M2, 8x
Field of View (°): 15 (est)
Acquisition Range (m): INA
Night: 1PN22M2 II channel, 6x
Field of View (°): 6 (est)
Acquisition Range (m): 800-1,000, based on light

VARIANTS
BRM-1: Baseline armored reconnaissance vehicle (BMP M1976/1) without smoke grenade launchers, added comms (R-130, R-014D telegraph), and Tall Mike radar but with four more passengers.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
73-mm HEAT-FS, PG-9
Maximum Aimed Range (m): 1,300
Max Effective Range (m):
Day: 800, 600 on the move
Night: 800
Armor Penetration (mm): 335 (RHA)

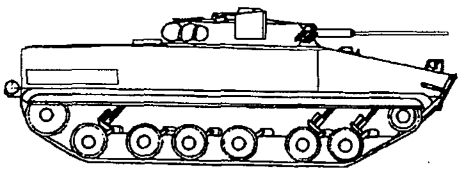
73-mm HEAT-FS, NFI
Maximum Aimed Range (m): 1,300
Max Effective Range (m):
Day: 1,000, 600 on the move
Night: 800-1,000
Armor Penetration (mm): >400 (RHA)

73-mm HE, OG-9
Maximum Aimed Range (m): 1,300
Max Effective Range (m):
Day: 1,300, 1,000 on the move
Night: 1,000
Armor penetration (mm): INA

Other Ammunition Types: 73-mm HE, OG-9M

NOTES
Derived from BMP-1, the vehicle has a 2-man turret and additional sensors. Two manportable SAM launchers are included. BMP-1 options fit BRM-1 and -1K. SENSORS: 1PN22M2 sight, 1D8 laser rangefinder, and Tall-Mike battlefield surveillance radar. Radar characteristics: operating band I (9.0 GHz); detection ranges 30 km personnel, 12 km vehicles. The Russian Alis or Sanoet thermal gunner's sight can be installed. Passengers may dismount from BRM-1K and will dismount from BRM-1 to form an alternate reconnaissance post.

Russian Combat Reconnaissance Vehicle BRM-3K



Weapons & Ammunition Types	Typical Combat Load
30-mm auto gun	**500**
HE-I & Frag-HE-T	340
APDS, APFSDS-T	160
7.62-mm coax MG	**2,000**

SYSTEM
Alternative Designations: Lynx, Rys
Date of Introduction: 1990
Proliferation: At least 1 country
Description:
Crew: 6
Combat Weight (mt): 19.6
Chassis Length Overall (m): 6.10
Height Overall (m): 2.65
Width Overall (m): 3.15
Ground Pressure (kg/cm^2): 0.62

Automotive Performance:
Engine Type: 500-hp Diesel
Cruising Range (km): 600
Speed (km/h):
Max Road: 70
Max Off-Road: 45
Average Cross-Country: 35
Max Swim: 10
Fording Depths (m): Amphibious

Radio: R-163-50U UHF, R-163-50K HF, R-163-10U (dismounts)

Protection:
Armor, Turret Front (mm): 30-35 mm (front glacis)
Applique Armor (mm): Yes on turret
Explosive Reactive Armor (mm): Available
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
Active Protective System: N/A
NBC Protection System: Collective
Smoke Equipment: 6 Smoke grenade launchers, VEESS

ARMAMENT
Main Armament:
Caliber, Type, Name: 30-mm automatic gun, 2A72
Rate of Fire: 350 rd/min (cyclic) in bursts
Loader Type: Dual-belt feed
Ready/Stowed Rounds: 500/ 0
Elevation (°): -5 to + 60
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm machinegun, PKT
Mount Type: Turret coax
Max Effective Range:
Day: 2,000 m
Night: 1,200-1,500 passive/2,000 active
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, in 2-10 round bursts

Firing Ports: 1 on each side

FIRE CONTROL
FCS Name: BPK-2-42
Main Gun Stabilization: 2-plane, 2E52-1
Rangefinder: Laser
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: BPK-2-42
Field of View (°): 8
Acquisition Range (m): 4,000 (est)
Night: 1PN61 II/IR sight
Field of View (°): INA
Acquisition Range (m): 1,200-1,500/3,000+ active IR
Commander Fire Main Gun: INA

VARIANTS
N/A

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
30-mm APDS
Maximum Aimed Range (m): 4,000 (est)
Max Effective Range (m):
Day: 2,500
Night: 1,200-1,500 passive/ 2,500 active
Tactical AA Range: 4,000
Armor Penetration (mm): 25 (RHA) at 1,500 m

30-mm APFSDS-T M929
Maximum Aimed Range (m): 4,000 (est)
Max Effective Range (m):
Day: 2,500+
Night: 1,200-1,500 passive/2,500+ active
Tactical AA Range: 4,000
Armor penetration (mm): 55 (RHA) at 1,000 m, 45 at 2,000 m

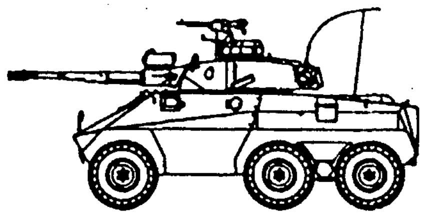
30-mm Frag-HE
Maximum Aimed Range (m): 4,000
Max Effective Range (m):
Day: 4,000
Night: 1,200-1,500 passive/ 3,000+ active
Tactical AA Range: 4,000
Armor Penetration (mm): INA

Other Ammunition Types: 30-mm HEI-T, AP-T

NOTES
BRM-3K is a variant of BMP-3 with a steel hull.

ONBOARD SENSORS: The 1PN71 thermal sight (3.7x/11.5x) has an acquisition range against tanks of 3.0 km. The 1D14 laser rangefinder (73x and 18x sights) has a day light only acquisition range of 10.0 km. The 1PN61 passive image intensifier night sight uses a laser illuminator. In the passive mode, the Generation II (7x) sight has a night acquisition range of 1.2-1.5 km. Using the active laser pulse illuminator, the acquisition range can be extended. Tall Mike Radar has an operating band I (9.0 GHz), and detection ranges: 3.0 km against personnel, 12.0 against moving vehicles.

Brazilian Armored Reconnaissance Vehicle EE-9



Weapons & Ammunition Types	Typical Combat Load
90-mm cannon	**44**
APFSDS-T	(est) 11
HEAT-T, HESH	11
HE-T	22
7.62-mm coax MG	**2,000**
.50 cal AA MG	**500**

SYSTEM
Alternative Designations: Cascavel IV
Date of Introduction: 1977
Proliferation: At least 18 countries (all variants)
Description:
Crew: 3
Troop Capacity: None
Combat Weight (mt): 13.4
Chassis Length Overall (m): 5.19
Height Overall (m): 2.36
Width Overall (m): 2.66
Drive Formula: 6 x 6

Automotive Performance:
Engine Type: 212-hp Diesel
Cruising Range (km): 880
Speed (km/h):Max Road: 100
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: N/A
Fording Depth (m): 1.0 unprepared

Radio: INA

Protection:
Armor, Turret Front (mm): 16
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: N/A
Smoke Equipment: 6 smoke grenade launchers

ARMAMENT
Main Armament:
Caliber, Type, Name: 90-mm gun, Engesa EC-90 (Cockerill-type)
Rate of Fire (rd/min): INA
Loader Type: Manual
Ready/Stowed Rounds: 24/20
Elevation (°): -8/+15
Fire on Move: INA

Auxiliary Weapons:
Caliber, Type, Name: 7.62-mm MG, INA
Mount Type: Coax
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: INA
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): INA

Caliber, Type, Name: .50 Cal M2 HB MG
Mount Type: Cupola
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 2,000
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): INA

ATGM Launcher: N/A
Firing Ports: N/A

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: N/A
Rangefinder: LV3 laser rangefinder
Infrared Searchlight: N/A
Sights w/Magnification:
Gunner:
Day: SS-123, 10x
Field of View (°): INA
Acquisition Range (m): INA
Night: SS-122 II channel, 5.6x
Field of View (°): INA
Acquisition Range (m): INA
Commander Fire Main Gun: No

VARIANTS
Cascavel I: Original vehicle had a US M36 37-mm gun turret.
Cascavel II: Variant with a French 90-mm gun from AML-90.
Cascavel III: Uses the 90-mmCockerill gun and new transmission.
Cascavel IV: Has a new engine and transmission, improved day and night optics with laser rangefinder, and .a 50 cal antiaircraft MG.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
90-mm APFSDS-T, Engequimica-produced
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 2,000+
Night: INA
Armor Penetration (mm): INA

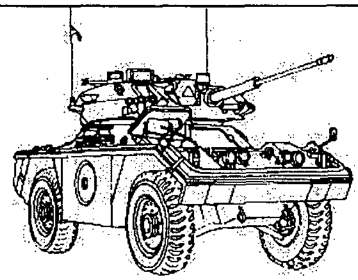
90-mm HE-T, Engequimica-produced
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 2,200
Night: INA
Armor Penetration (mm): INA

Other Ammunition Types: HEAT-T, HESH-T, Smoke, Cannister

NOTES
Other ammunition maximum effective ranges are (m): HEAT-T - 1,500, HESH-T - 800.

British Armored Reconnaissance Vehicle Fox

	Weapons & Ammunition	Typical



Types	Combat Load
30-mm auto-cannon	**99**
HEI-T,	(est) 66
APDS-T, APSE-T	33
7.62-mm coax MG	**2,600**

SYSTEM
Alternative Designations: FV721
Date of Introduction: 1973
Proliferation: At least 3 countries

Description:
Crew: 3
Troop Capacity: 0
Combat Weight (mt): 6.12
Chassis Length Overall (m): 4.17
Height Overall (m): 2.20
Width Overall (m): 2.13
Ground Pressure (kg/cm^2): INA
Drive Formula: 4 x 4

Automotive Performance:
Engine Type: 190-hp Gasoline
Cruising Range (km): 434
Speed (km/h):
Max Road: 104
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 5.23
Fording Depth (m): 1.0 Unprepared

Radio: INA

Protection:
Armor, Turret Front (mm): Resistant to heavy MG (NFI)
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: N/A
Smoke Equipment: 2 x 4-barrel smoke grenade launchers

ARMAMENT
Main Armament:
Caliber, Type, Name: 30-mm auto-cannon, Rarden L21
Rate of Fire (rd/min): 80-90 cyclic (1-6 round bursts)
Loader Type: Feed tray, manual clip-fed (3-round clips)
Ready/Stowed Rounds: INA
Elevation (°): -14/+40
Fire on Move: INA

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm machinegun L7A2
Mount Type: Coax
Maximum Aimed Range (m): INA
Max Effective Range (m): INA
Fire on Move: Yes
Rate of Fire (rd/min): INA

ATGM Launcher: N/A
Firing Ports: None

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: N/A
Sights w/Magnification:
Gunner:
Day: SPAV L2A1, 5.8/1.6 x
Field of View (°): 8/28
Acquisition Range (m): INA
Night: L2A1, II sight
Field of View (°): INA
Acquisition Range (m): INA

Commander Fire Main Gun: No

VARIANTS
None of the variants have been fielded.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
30-mm APDS-T, L14A2
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,000
Night: INA
Tactical AA Range: INA
Armor Penetration (mm): 40 (RHA, 45°) at 1,500 meters

Caliber, Type, Name: 30-mm HE-T, L13A1
Maximum Aimed Range (m): 2,000
Max Effective Range (m): INA
Tactical AA Range: INA
Armor Penetration (mm): N/A

Other Ammunition Types: APSE-T (AP Secondary Effects-T L5A2). The gun can fire the KCB (Oerlikon) family of munitions.

NOTES
These vehicles have been phased out of British service.

Chapter 4
Tanks/Assault Vehicles

The lethality and variety of weapons available to armored, mechanized, and infantry forces for the close fight require a continued and expanded use of heavily armored fighting vehicles (AFVs). This chapter provides a representative sampling of AFVs in use today and designed for combat assault. The selection is not comprehensive, rather reflects a mix of systems currently available for the OPFOR and likely to be encountered in varying levels of conflict. The selection is also used to highlight trends within this field of weapons.

Vehicles used for combat assault in this Guide are divided into two categories—*main battle tanks and light tanks/assault vehicles*. Tanks are tracked, heavily armored vehicles with guns of generally 75 mm or more. Among modern trends in AFVs are: increased variety of systems worldwide, and a wider application of these systems for varied roles and missions on the battlefield. As a result, technology sharing and proliferation of upgrade packages have blurred lines among vehicles used for assault, antiarmor, combat reconnaissance and fire support missions. Another trend is increased weight for all types of armored vehicles. With heavier armor protection packages, higher-output engines and larger weapons, a significant proportion of medium tanks have grown into the heavy tank weight category. Therefore, the term ***main battle tank*** is more relevant than previous weight categories.

There are still ***light tanks*** on the battlefield, although increased armor and gun size on light armored fighting vehicles such as infantry fighting vehicles and armored reconnaissance vehicles have blurred lines of distinction. A number of AFVs, such as the British Scorpion and French AMX-13 can be characterized as reconnaissance vehicles, tank destroyers, fire support vehicles, or assault vehicles; but they have tracks, armor protection, and guns of 60 mm or greater. Thus, they can also be used for light tank missions. The term ***assault vehicle*** currently represents a narrow category of older vehicles used by (former) Soviet forces - medium-armored vehicles with medium-heavy guns and no turrets. None of these vehicles were selected for this initial publication. Some representative systems will be included in the next iteration. With blurring of lines among roles and missions for heavier LAFVs and light tanks, the term *assault vehicle* will likely broaden to reflect a variety of modern programs for light - medium armored vehicles with medium to heavy guns, for use in the assault role.

Two notable trends for vehicles in this chapter are a reflection of increasing systems costs and declines or leveling of military budgets - development of variants off of established systems, and use of equipment/packages to extend the use life of systems and enhance their effectiveness. As a result, seemingly old and out-of-date tanks, some of which pre-date World War II, can be a threat to modern armored and mechanized forces. The WEG highlights a variety of upgrades as well as limitations for selected tanks. Systems-related trends can be divided among mobility, survivability, and lethality, as noted on the data sheets.

To improve mobility and compensate for weight increases, many forces have replaced older engines with more powerful diesel engines. Swim capability is limited to a few light tanks.

Within the area of survivability, the most obvious consideration is increasing armor protection levels. A prominent trend is the application of additional armor, such as plate armor or panels on turrets, side-skirts over tracks, and addition of explosive reactive armor (ERA). Additional protection measures include use of entrenching blades for self-emplacement, mine-clearing plows and rollers, nuclear, biological and chemical (NBC) protection, vehicle smoke emission systems, and smoke grenade launchers. To complement these systems are sensors such as mine detectors, laser warning receivers, and radar warning receivers. A trend receiving increasing attention is the use of active measures: electro-optical countermeasures, such as infrared jammers, and active protection systems (also known as defensive aides suites) designed to intercept incoming projectiles and destroy them prior to impact.

The area of lethality has seen a variety of upgrades, including: gun replacement, improved stabilization and fire control systems, additional weapons such as antitank guided missile systems, and improved ammunition. Critical parameters include fire on the move capability, which can be linked to stabilization, rate of fire, integrated sights, acquisition ranges, and weapon range. Note, because weapon range is really a function of sights, gun precision, the type of mount, and specific round ballistics, the WEG will incorporate those factors in the round data, as maximum aimed range. That figure conforms to the OPFOR tactics and accounts for technical capabilities (see Glossary). Maximum effective range is also included (see Glossary).

The WEG notes a variety of new ammunition natures, such as electronically fuzed tank rounds for use against helicopters, and OPFOR availability of western-style HEAT-multipurpose rounds, which can be used as both antitank and antipersonnel rounds, for greater flexibility and lethality. For some systems, the ammunition mix could be determined or estimated. For others, that data was not available. Within each category, the specific round mix will depend on tactical considerations, comparative lethality and the intended targets. A general rule for OPFOR is that tanks will have approximately 50% antitank rounds and 50% rounds for use against soft targets. Because of the relative increase in protection against HEAT rounds vs kinetic energy rounds, mix estimates reflect a bias toward KE rounds. The term *stowed rounds* does not mean rounds which are not in the tank's autoloader. Rounds in ready reach are ready rounds. Stowed rounds are those which are in compartments away from the gunner's or loader's positions, requiring a slower than normal rate of fire (see Glossary). In calculating tank rounds, the figure does not include the tactical possibility of adding an additional round in the breach.

Secondary arms continue to play an important role for OPFOR tanks, because their use permits the main gun to focus fires more on heavy and area targets. Tankers will fire main guns at hovering or slow-flying aircraft; however, the more likely weapon is the antiaircraft machinegun. Similarly, OPFOR tanks will fire main guns at personnel and other soft targets as required; but the more efficient weapon for targets at close range is the coaxial machinegun.

Questions and comments on data listed in this chapter should be addressed to:

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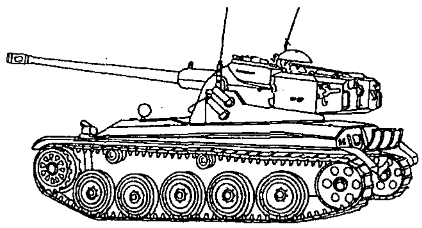
French Light Tank AMX-13



AMX-13 Model 51/75 mm Gun

Weapons & Ammunition Types	Typical Combat Load
90-mm rifled gun	
APFSDS-T	34
HEAT	
HE	
Cannister	
7.62-mm coax MG	3,600

SYSTEM
Alternative Designations: AMX-13/90
Date of Introduction: 1966
Proliferation: At least 15 countries
Description:
Crew: 3
Combat Weight (mt): 15.0
Chassis Length Overall (m): 4.88
Height Overall (m): 2.28
Width Overall (m): 2.51
Ground Pressure (kg/cm^2): 0.74

Automotive Performance:
Engine Type: 250-hp Gasoline
Cruising Range (km): 350
Speed (km/h):
Max Road: 60
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: N/A
Fording Depths (m): 0.6 unprepared, 2.1 with snorkel

Radio: TR-VP118 and intercom

Protection:
Armor, Turret Front (mm): 25 at 45° impact angle
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: N/A
Smoke Equipment: 2 smoke grenade launchers each side of turret

ARMAMENT
Main Armaments:
Caliber, Type, Name: 90-mm rifled gun CN-90-F3
Rate of Fire (rd/min): INA
Loader Type: Autoloader and manual
Ready/Stowed Rounds: 10 in autoloader, 11/13 in hull
Elevation (°): -5.5/+12.5
Fire on Move: N/A

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x51) MG, AA52
Mount Type: Turret coax
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: INA
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): INA

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: L862, 7.5x and 8x
Field of View (°): INA
Acquisition Range (m): INA
Night: OB-11-A, 5x
Field of View (°): INA
Acquisition Range (m): 800-1,000
Commander Fire Main Gun: No

VARIANTS
AMX-13 Model 51: Original tank destroyer/recon vehicle, Model 51, w/75-mm gun. Many variants and upgrades have diesel engines and a 7.62-mm AA MG. Two versions were fitted with 2 x SS-11 or 3 x HOT ATGM launchers
AMX-13/90: This is the variant portrayed on this data sheet.
AMX-13/105: Variant with a GIAT 105G1 105-mm gun.
AMX-13 CD Model 55: Armored recovery variant.
AMX-13 DCA: Air defense variant with twin 30-mm guns.
AMX-13 with LAR: Multiple Rocket Launcher System.
AMX 105-mm Mk 61: Self-propelled howitzer variant.
AMX F3: 155-mm self-propelled gun.
AMX-VCI: Variant used as an APC.

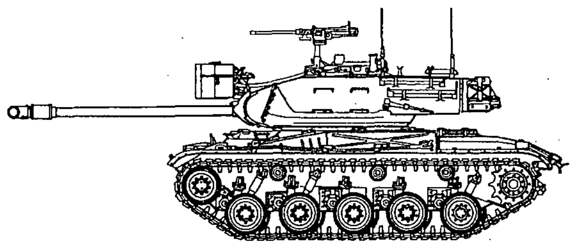
MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
90-mm APFSDS-T, NFI
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 2,000
Night: 800-1,000
Armor Penetration (mm): INA

90-mm HEAT, NFI
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 1,000
Night: N/A
Armor Penetration (mm): 160 (RHA) at 60° impact angle

Other Ammunition Types: HE, Cannister, Smoke

NOTES
Israeli EL-OP thermal sights are available for use on the tank.

US Light Tank M41A3



Weapons & Ammunition Types	Typical Combat Load
76-mm rifled gun M32	**65**
APDS-T/APFSDS-T	20
HEAT -T	20
Frag-HE	20
Cannister	5
7.62-mm coax MG	**5,000**
12.7-mm AA MG	**2,175**

SYSTEM
Alternative Designations: Walker Tank, Walker Bulldog
Date of Introduction: 1951
Proliferation: At least 18 countries
Description:
Crew: 4
Combat Weight (mt): 23.5
Chassis Length Overall (m): 5.82
Height Overall (m): 2.73
Width Overall (m): 3.20
Ground Pressure (kg/cm²): 0.72

Automotive Performance:
Engine Type: 500-hp Gasoline
Cruising Range (km): 161
Speed (km/h):
Max Road: 72
Max Off-Road: 48
Average Cross-Country: 40
Max Swim: N/A
Fording Depths (m): 1.0 Unprepared, 2.4 prepared

Radio: INA

Protection:
Armor, Turret Front (mm): 38
Applique Armor (mm): Available
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: N/A
Smoke Equipment: N/A

ARMAMENT
Main Armament:
Caliber, Type, Name: 76-mm rifled gun M32
Rate of Fire (rd/min): INA
Loader Type: Manual
Ready/Stowed Rounds: INA
Elevation (°): -9.75/ +19.75
Fire on Move: No

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x51) MG, M9194E1
Mount Type: Turret coax
Maximum Aimed Range (m):
Max Effective Range (m):
Day: INA
Night: N/A
Fire on Move: Yes
Rate of Fire: INA

Caliber, Type, Name: .50 (12.7 x 99) AA machinegun, M2HB
Mount Type: Cupola AA mount
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 2,000
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): 450-550

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: N/A
Rangefinder: N/A
Infrared Searchlight: Available
Sights w/Magnification:
Gunner:
Day: M97A1 and M20A1
Field of View (°): INA
Acquisition Range (m): INA
Night: Available
Commander Fire Main Gun: No

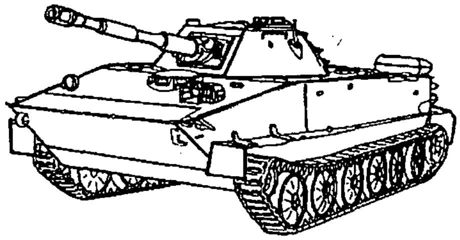
VARIANTS
M41 DK-1: Danish variant with diesel engine and LRF-based fire control. Other upgrades are side skirts, thermal sights, NBC protection, smoke grenade launchers and 7.62-mm AA MG.
Brazilian M41: Upgrades are similar to DK-1 except for AA MG and change to 90-mm gun using Cockerill Mk III ammunition.
Uruguayan M41: M41A3 fitted with Cockerill Mk III gun.
YUNG HU: Taiwanese upgrade with diesel engine.
M42/Duster: Air defense gun system with twin 40-mm AA cannon.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
76-mm APFSDS-T, AAI M464
Maximum Aimed Range (m): INA
Max Effective Range (m): INA
Armor Penetration (mm): NATO triple heavy (57°) at 1000 m

Other Ammunition Types: M33A1 and A2 APDS-T, M319 and M339 AP-T, M496 HEAT-T, HE, Smoke (WP), M363 cannister

NOTES
German Atlas offers the MOLF 1-plane stabilized laser rangefinder fire control system and retrofit kit The FCS includes a thermal night sight. Israeli EL-OP offers a FCS for the system. Maximum range for the canister round is 155 meters.

Russian Amphibious Tank PT-76B



Weapons & Ammunition Types	Typical Combat Load
76-mm rifled gun D-56	**40**
HVAP, AP-T/API-T	10
HEAT	10
Frag-HE	20
7.62-mm coax MG	**1,000**

SYSTEM
Alternative Designations: INA
Date of Introduction: 1952
Proliferation: At least 21 countries
Description:
Crew: 3
Combat Weight (mt): 14.0
Chassis Length Overall (m): 6.91
Height Overall (m): 2.26
Width Overall (m): 3.14
Ground Pressure (kg/cm^2): 0.46

Automotive Performance:
Engine Type: 240-hp Diesel
Cruising Range (km): 260
Speed (km/h):
Max Road: 44
Max Off-Road: INA
Average Cross-Country: 25
Max Swim: 10
Fording Depth (m): Amphibious

Radio: R-123
Protection:
Armor, Turret Front (mm): 20
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: N/A
Smoke Equipment: VEESS

ARMAMENT
Main Armament:
Caliber, Type, Name: 76-mm rifled gun D-56B
Rate of Fire (rd/min): 6-8
Loader Type: Manual
Ready/Stowed Rounds: INA
Elevation (°): -4/+30
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x54R) machinegun PKT
Mount Type: Coax
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,000/400-500 on the move
Night: 600
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, 2-10 round bursts

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: 2-plane
Rangefinder: N/A
Infrared Searchlight: Available
Sights w/Magnification:
Gunner: TShK-66
Day:
Field of View (°): INA
Acquisition Range (m): 4,000
Night: TVN-28 IR Available
Field of View (°): INA
Acquisition Range (m): 600
Commander Fire Main Gun: No

VARIANTS
Polish PT-76: Variant with a separate commander's hatch and 12.7-mm MG.

Type 63: Chinese variant with a new turret, 85-mm gun, and 12.7-mm AA MG.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
76-mm HVAP-T, BM-354P
Maximum Aimed Range (m): 1,060
Max Effective Range (m):
Day: 650
Night: 600
Armor Penetration (mm): 127 at muzzle, 50 at 1,000 m

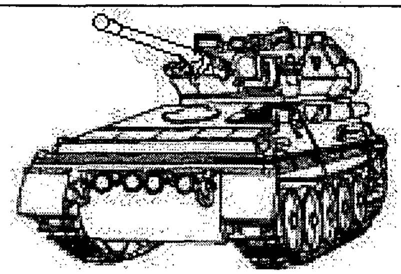
76-mm HEAT, BK-350M
Maximum Aimed Range (m): 1,000
Max Effective Range (m):
Day: 650
Night: 600
Armor Penetration (mm): 280 to max range

76-mm, Frag-HE, OF-350
Maximum Aimed Range (m): 4,000
Max Effective Range (m):
Day: INA
Night: 600
Armor Penetration (mm): INA

Other Ammunition Types: 76-mm AP-T, BR-350 API-T

NOTES
Original PT-76 was produced in limited numbers with a non-stabilized main gun. Some PT-76s are augmented with 12.7-mm AA MGs. Israel offers an upgrade package with a 90-mm gun, LRF fire control and a 300-hp engine.

British Combat Reconnaissance Vehicle, Tracked Scorpion



Weapons & Ammunition Types	Typical Combat Load
76-mm rifled gun	40
HESH	
HE	
Cannister	
7.62-mm coax MG	3,600

SYSTEM
Alternative Designations: FV101
Date of Introduction: 1972
Proliferation: At least 18 countries
Description:
Crew: 3
Combat Weight (mt): 8.07
Chassis Length Overall (m): 4.79
Height Overall (m): 2.10
Width Overall (m): 2.24
Ground Pressure (kg/cm^2): 0.36

Automotive Performance:
Engine Type: 190-hp Gasoline
Cruising Range (km): 650
Speed (km/h):
Max Road: 80
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 4/6 with propeller
Fording Depth (m): 1.07, amphibious

Radio: INA

Protection:
Armor, Turret Front (mm): Against 14.5-mm projectiles
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: N/A
Self-Entrenching Blade: N/A
NBC Protection System: Yes
Smoke Equipment: 4 smoke grenade launchers each side of turret

ARMAMENT
Main Armament:
Caliber, Type, Name: 76-mm rifled gun L23A1
Rate of Fire (rd/min): 6
Loader Type: INA
Ready/Stowed Rounds: INA
Elevation (°): -10/ +35
Fire on Move: N/A

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x51) MG, L8A1
Mount Type: Turret coax
Maximum Aimed Range (m): INA
Max Effective Range (m): INA
Fire on Move: Yes
Rate of Fire (rd/min): INA

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: N/A
Rangefinder: Laser rangefinder
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: Barr and Stroud Tank Laser Sight, 10x
Field of View (°): INA
Acquisition Range (m): 2,200
Night: GEC Sensors SS100, II, x5.8/1.6
Field of View (°): 8/28
Acquisition Range (m): INA
Commander Fire Main Gun: No

VARIANTS
Scorpion 90: Variant with a 90-mm Cockerill Mk III gun.

A number of vehicles use the same Alvis chassis. They include the **Scimitar** armored reconnaissance vehicle, **Striker** armored ATGM launcher vehicle, **Spartan** armored personnel carrier or Milan ATGM launcher, **Stormer** modernized APC, **Samaritan** armored ambulance, and **Saber** modernized reconnaissance vehicle.

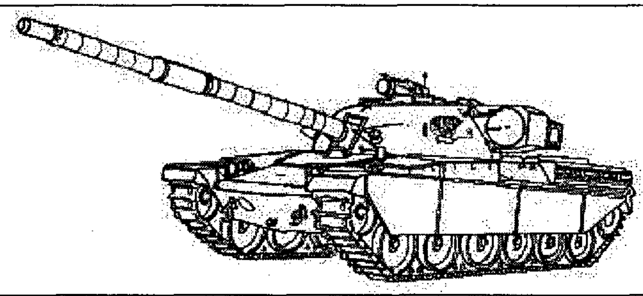
MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
76-mm HESH, L29
Maximum Aimed Range (m): 2,200
Max Effective Range (m): INA
Armor Penetration (mm): INA

Other Ammunition Types: L24A1/2 HE (max effective range--indirect fire: 5,000 meters), L33A1 Cannister (max effective range: 100 meters), L32A5 Smoke (BE), L42 Illumination

NOTES
As a reflection of the vehicle's suitability for a variety of roles, in recent times it is referred to as an armored reconnaissance vehicle or combat vehicle reconnaissance (tracked)--CVR (T).

A British upgrade program includes a diesel engine, thermal sights, and secure communications. The Tank Laser Sight and Avimo LV10 Day/Night LRF sight can accept a thermal channel. Thermal sights are available for use on the tank.

British Main Battle Tank Chieftain Mk 5



Weapons & Ammunition Types	Typical Combat Load
120-mm rifled gun	**64**
APFSDS-T	20
HESH	44
7.62-mm MG	**6,200**
--Coaxial and Stowed	**6,000**
--Cupola AA MG	**200**

SYSTEM
Alternative Designations: FV 4201
Date of Introduction: 1967 Original Chieftain
Proliferation: At least 6 countries
Description:
Crew: 4
Combat Weight (mt): 55.00
Chassis Length Overall (m): 7.48
Height Overall (m): 2.90
Width Overall (m): 3.51
Ground Pressure (kg/cm^2): 0.90

Automotive Performance:
Engine Type: 750-hp Diesel
Cruising Range (km): 400-500
Speed (km/h):
Max Road: 48
Max Off-Road: INA
Average Cross-Country: 30
Max Swim: N/A
Fording Depths (m): 1.1 Unprepared

Radio: C42/Larkspur VHF

Protection:
Armor, Turret Front (mm): 300 (RHA)
Applique Armor (mm): ROMOR applique on turret, side skirts
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: Plow variant, and AVLB/engineer variant
Self-Entrenching Blade: No
NBC Protection System: Yes
Smoke Equipment: Smoke grenade launchers (6 each side of turret)

ARMAMENT
Main Armaments:
Caliber, Type, Name: 120-mm rifled gun, L11A5
Rate of Fire (rd/min): 8-10 first minute/6 sustained
Loader Type: Separate-loading manual
Ready/Stowed Rounds: INA
Elevation (°): -10 to +20
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 51) Machine gun L8A1
Mount Type: Turret Coax
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 800
Night: INA
Fire on Move: Yes
Rate of Fire: INA

Caliber, Type, Name: 7.62-mm (7.62x 51) AA Machine gun L37A1
Mount Type: Cupola
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 800
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): INA

ATGM Launcher: N/A

FIRE CONTROL
FCS Name: Improved Fire Control System (IFCS)
Main Gun Stabilization: 2-plane
Rangefinder: Laser, Nd-Yag
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: Barr and Stroud Tank Laser Sight (TLS), 8x
Field of View (°): 10
Acquisition Range (m): 5,000
Night: 1R18 Thermal sight, 3x
Field of View (°): INA
Acquisition Range (m): INA

Commander Fire Main Gun: INA

VARIANTS
Mk 5: Final production variant, with a new engine and NBC system, modified auxiliary weapons and sights. Mk 6-11 are upgrades to earlier models, with addition of IFCS. Mk 12 added ROMOR (aka: Stillbrew) spaced armor boxes. Mk 11 and Mk 12 have Thermal Observation and Gunnery Sight (TOGS).

A variety of support vehicles were developed from the tank. They include recovery vehicles, AVLB, dozer, mineclearer, air defense and 155-mm SP artillery systems.

Khalid/Shir 1: Jordanian variant which has chassis, turret and weaponry of the Chieftain, but which incorporates engine and running gear upgrades of Challenger 1. The fire control has seen a number of improvements, including a new ballistic computer.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
120-mm APFSDS-T, L23A1
Maximum Aimed Range (m): 5,000
Max Effective Range (m):
Day: 3,000
Night: INA
Armor Penetration (mm): INA

British Main Battle Tank Chieftain Mk 5 continued

120-mm High-Explosive Squash-Head (HESH), L31 Maximum Aimed Range (m): 5,000 Max Effective Range (m): Day: 3,000 Night: INA Armor Penetration (mm): INA	**Other Ammunition Types:** L15 APDS, L34 WP Smoke

NOTES
Early Chieftains and some later modified tanks mount the 50. Cal M2HB machinegun over the main gun as a ranging gun. Iran and Kuwait retained the .50 Cal MG.

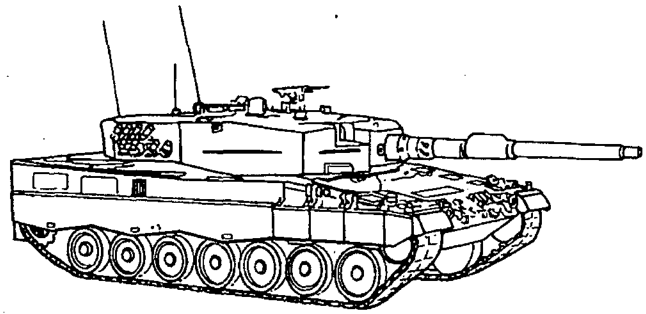
The HESH round is used for antitank chemical-energy (CE) antiarmor missions, and for HE effects against personnel and materiel.

The Iranians claim to employ a snorkel system on Chieftain, for fording to 5 meters depth.

A variety of fire control systems and thermal sights are available for Chieftain. At 324 Chieftains have been upgraded with the Barr and Stroud TOGS thermal sight system. The 1R26 thermal camera can be used with the 1R18 thermal night sight. It has wide (13.6°) and narrow (4.75°) fields of view, and is compatible with TOGS format. GEC Sensors offers a long list of sights including: Multisensors Platform, Tank Thermal Sensor, and SS100/110 thermal night sight. Marconi, Nanoquest, and Pilkington offer day and night sights for the Chieftain.

Charm Armament upgrade program, with the 120-mm L30 gun incorporated in Challenger 1, is available for Chieftain modification programs.

German Main Battle Tank Leopard 2



Weapons & Ammunition Types	Typical Combat Load
120-mm smoothbore gun	42
APFSDS-T	
HEAT-MP-T	
7.62-mm machineguns	4750
--Coaxial	2000
--Cupola MG/stowed	2750

SYSTEM
Alternative Designations: Swiss Pz 87, Swedish Strv 121
Date of Introduction: 1979
Proliferation: At least 7 countries
Description:
Crew: 4
Combat Weight (mt): 55.15
Chassis Length Overall (m): 7.69
Height Overall (m): 2.79
Width Overall (m): 3.70
Ground Pressure (kg/cm^2): 0.83

Automotive Performance:
Engine Type: 1,500-hp Diesel
Cruising Range (km): 550
Speed (km/h):
Max Road: 72
Max Off-Road: 45
Average Cross-Country: 40
Max Swim: N/A
Fording Depths (m): 1.0 Unprepared, 4.0 with snorkel

Radio: INA

Protection:
Armor, Turret Front (mm): 700 KE/1,000 against HEAT rounds
Applique Armor (mm): Track skirt
Explosive Reactive Armor (mm): N/A
Active Protective System: Galix (See note Strv 122)
Mineclearing Equipment: No
Self-Entrenching Blade: No
NBC Protection System: Yes
Smoke Equipment: Smoke grenade launchers, 8 each side of turret

ARMAMENT
Main Armaments:
Caliber, Type, Name: 120-mm smoothbore gun M256
Rate of Fire (rd/min): INA
Loader Type: Manual
Ready/Stowed Rounds: 15/27
Elevation (°): -9 to +20
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 51) Machinegun MG3A1
Mount Type: Turret Coax
Maximum Aimed Range(m): INA
Max Effective Range (m):
Day: INA
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): 1,200

Caliber, Type, Name: 7.62-mm (7.62x 51) Machinegun MG3A1
Mount Type: Turret Cupola
Maximum Aimed Range(m): INA
Max Effective Range (m):
Day: INA
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min): 1,200
ATGM Launcher: N/A

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: WNA-H22, 2-plane
Rangefinder: Laser neodymium
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: Krupp-Atlas EMES-15, 12x / FERO Z18 secondary, 8x
Field of View (°): 5/10
Acquisition Range (m): INA
Night: Zeiss thermal imager
Field of View (°): INA
Acquisition Range (m): INA
Commander Fire Main Gun: Yes

VARIANTS
A variety of MBT variants from 2A1 to 2A4 denote minor changes, as well as FCS upgrades. Combat support variants include an armored recovery vehicle.

Pz87: Swiss variant with indigenous machineguns, communications and FCS, and improved NBC equipment.

Dutch Leopard 2: Uses indigenous equipment as noted above.

Leopard 2A5/Leopard 2 (Improved): Recent upgrade with spaced armor added to turret front, and increased armor on hull and side skirts. Other improvements include improved stabilization, suspension, navigation, fire control, and hatch design.

Strv 122: Swedish-licensed variant resembling 2A5 with an indigenous turret and other upgrades. The tank features French Galix active protection system and improved command and control. Sweden developed an HE-T round designed to range 2,000 meters or more for its Leopard-2 and Strv-122 tanks. With additional armor, Strv 122 will weigh 62 mt.

German Main Battle Tank Leopard 2 continued

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
120-mm APFSDS-T, DM43
Maximum Aimed Range(m): 3,500
Max Effective Range (m):
Day: INA
Night: INA
Armor Penetration (mm): 450 at 2,000 meters

120-mm APFSDS-T, US Olin GD120
Maximum Aimed Range(m): 3,500
Max Effective Range (m):
Day: 3,000
Night: INA
Armor Penetration (mm): 520 at 2,000 meters

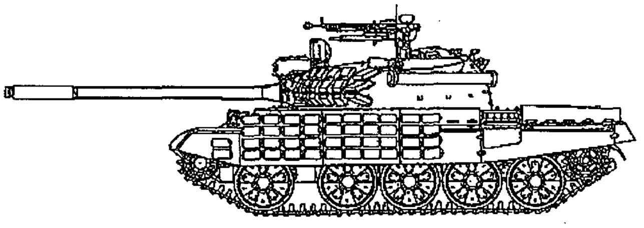
120-mm HEAT-MP-T, DM-12A1/US Olin M830
Maximum Aimed Range(m): INA
Max Effective Range (m):
Day: 2,500
Night: INA
Armor Penetration (mm): INA

Other Ammunition Types: US-produced M829, M829A1 APFSDS-T; US M830A1 HEAT-MP-T (MPAT), GE DM12A1 (US copy M830) HEAT-MP-T (MPAT)

NOTES
A variety of upgrade programs and options are available for the Leopard 2. These include the Atlas Elektronik Vehicle Integrated Command and Information System (IFIS), a digital command and information system.

A new longer gun barrel (L55 gun barrel, 1.30 meters longer) is available. It permits effective use of a new APFSDS-T round, DM53 (LKE II), with a longer rod penetrator, and which is under development. The German Army has decided not to buy the DM43 APFSDS-T round (aka: LKE 1), rather to wait and upgrade to the DM53.

Russian Main Battle Tank T-55AMV



Weapons & Ammunition Types	Typical Combat Load
100-mm rifled gun	(mix est) **43**
APFSDS-T	14
HEAT	3
Frag-HE	21
ATGM	5
7.62-mm coax PKT MG	**1,250**
12.7-mm AA MG	**500**

SYSTEM
Alternative Designations: INA
Date of Introduction: 1983
Proliferation: At least 3 countries
Description:
Crew: 4
Combat Weight (mt): 40.5
Chassis Length Overall (m): 6.20
Height Overall (m): 2.32
Width Overall (m): 3.60
Ground Pressure (kg/cm^2): 0.89

Automotive Performance:
Engine Type: 620-690 hp Diesel
Cruising Range (km): 390/600 with extra tanks
Speed (km/h):
Max Road: 50
Max Off-Road: 35
Average Cross-Country: 25
Max Swim: N/A
Fording Depths (m): 1.4 Unprepared, 5.5 with snorkel

Radio: R-173, R-173P, R-124 intercom

Protection:
Armor, Turret Front (mm): 200 (base T-55 armor)
Applique Armor (mm): Rubber screens and box armor
Explosive Reactive Armor (mm): 1st Gen raises to KE/700-900 against HEAT; 2nd Gen raises to 450-480 KE/700-900 HEAT
Active Protective System: Russian Drozd APS available
Mineclearing Equipment: Roller-plow set, and plows available
Self-Entrenching Blade: No
NBC Protection System: Yes
Smoke Equipment: Smoke grenade launchers (4x 81-mm each side of turret), and 24 grenades. Vehicle engine exhaust smoke system

ARMAMENT
Main Armaments:
Caliber, Type, Name: 100-mm rifled gun, D-10T2S
Rate of Fire (rd/min): 5-7
Loader Type: Manual
Ready/Stowed Rounds: INA
Elevation (°): -5 to +18
Fire on Move: Yes (gun rounds only--ATGMs require a short halt)

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) Machinegun PKT-T
Mount Type: Turret coax
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 800
Night: 800
Fire on Move: Yes
Rate of Fire (rd/min): 250 rpm practical, 800 cyclic, 2-10 rd bursts

Caliber, Type, Name: 12.7-mm (12.7x108) AA MG DShKM
Mount Type: Turret top
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,500
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 80-100 practical, 600 cyclic, 2-10 rd bursts

ATGM Launcher :
Name: D-10T2S gun
Launch Method: Gun-launched
Guidance: SACLOS, Infrared laser-beam rider
Command Link: Encoded laser-beam
Launcher Dismountable: No

FIRE CONTROL
FCS Name: Volna
Main Gun Stabilization: M1 Tsiklon 2-plane
Rangefinder: KDT-2 Laser
Infrared Searchlight: L-4
Sights w/Magnification:
Gunner:
Day: TShSM-32PV, 3.5x and 7x
Field of View (°): 18 and 8
Acquisition Range (m): 4,000
Night: 1K13
Field of View (°): INA
Acquisition Range (m): 800-1,300, gun rounds only
Commander Fire Main Gun: No

VARIANTS
More than a dozen countries have produced upgraded T-55 variants with similar capabilities in protection and lethality. Many countries have upgraded to a larger main gun.

T-55AMV is derived from a line of variants of T-55 MBT. **T-55A** added an NBC protection system. **T-55M** added the Volna fire control system (with ATGM launcher), improved gun stabilization and sights, improved engine, new radio, and increased protection. That included side skirts, smoke grenade launchers, applique armor, and fire protection. **T-55AM** added bra armor, an armor band around the turret for 180° coverage. The -AMV upgrade means substitution of ERA for the bra armor. Variants ending with **-1** denote replacement of the engine w/V-46 engine from T-72 MBT.
The Ukraine and Syria will upgrade to the T-55AMV standard.

Russian Main Battle Tank T-55AMV continued

T-55AM2B: Czech version of T-55AMV with Kladivo fire control.
T-55AM2: Variant does not have ATGM capability or Volna FCS.
T-55AM2P: Polish version of T-55AMV but with Merida FCS.
T-55AMD: Variant with the Drozd APS instead of ERA.
T-55AD Drozd: Variant with Drozd but not Volna FCS and ERA.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
100-mm BM-8 Russian
Maximum Aimed Range (m): 2,500
Max Effective Range (m):
Day: 1,500
Night: 800-1,300
Armor Penetration (mm): 200 at 1,000 meters

100-mm APFSDS-T, BM-25
Maximum Aimed Range (m): 2,500
Max Effective Range (m):
Day: INA
Night: 800-1,300
Armor Penetration (mm): INA

100-mm APFSDS-T, BM-412M, Romanian
Maximum Aimed Range (m): 2,500
Max Effective Range (m):
Day: 2, 000+ (est)
Night: 800-1,300
Armor Penetration (mm): 418 at 2,000 m, 380 at 3,000 m

100-mm APFSDS-T, M1000, Belgian
Maximum Aimed Range (m): 2,500
Max Effective Range (m):
Day: 2,500 (est)
Night: 800-1,300
Armor Penetration (mm): NATO triple heavy target, 4,500 m

100-mm HEAT, BK-17
Maximum Aimed Range (m): 2,500
Max Effective Range (m):
Day: 1,000 (est)
Night: 800-1,000 (est)
Armor Penetration (mm): 380

100-mm Frag-HE, OF-32
Maximum Aimed Range (m): 4,000
Max Effective Range (m):
Day: <2,500
Night: 800-1,300
Armor Penetration (mm): INA

Other Ammunition Types: A variety of other rounds within the range noted above are available. They include the GIAT NR 322/ NR 352 APFSDS-T and Slovak JPrSv AP-T with ranges beyond 2,000 m.

Antitank Guided Missiles:
Name: AT-10/BASTION
Warhead Type: Shaped charge (HEAT)
Armor Penetration (mm): 650 (RHA)
Range (m): 4,000 (day only, see NOTES)

Name: AT-10 Improved
Warhead Type: Tandem shaped charge
Armor Penetration (mm): 700 (RHA) behind ERA
Range (m): 4,000 (day only, see NOTES)

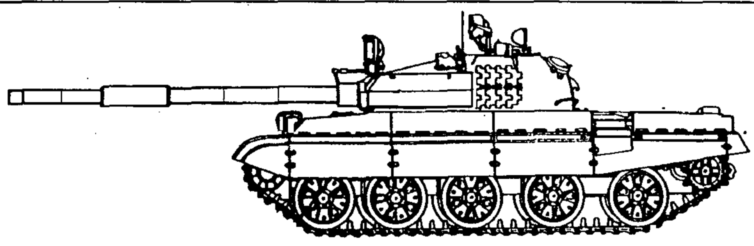
NOTES
The 1K13 sight is both night sight and ATGM launcher sight; however, it cannot be used for both functions simultaneously.

T-55s with "bra armor", semi-circular add-on armor, have turret protection increased to 330 mm (KE) and 400-450 mm (CE). Other improvements available include a hull bottom reinforced against mines, better engines, rubber track pads, and a thermal sleeve for the gun.

Optional sights and fire control systems include the Israeli El-Op Red Tiger and Matador FCS, Swedish NobelTech T-series sight, and German Atlas MOLF. The Serbian SUV-T55A FCS, British Marconi Digital FCS, South African Tiger, and Belgian SABCA Titan offer upgraded function. One of the best is the Slovenian EFCS-3 integrated FCS.

A variety of thermal sights is available. They include the Russian/French ALIS and Namut-type sight from Peleng. There are thermal sights available for installation which permit night launch of ATGMs.

Russian Main Battle Tank T-62M



Weapons & Ammunition Types	Typical Combat Load
115-mm rifled gun	(mix est) **40**
APFSDS-T	12
HEAT	3
Frag-HE	20
ATGM	5
7.62-mm coax PKT MG	**2,500**

SYSTEM
Alternative Designations: INA
Date of Introduction: 1983
Proliferation: At least 1 country

Description:
Crew: 4
Combat Weight (mt): 41.5
Chassis Length Overall (m): 6.63
Height Overall (m): 2.4
Width Overall (m): 3.52
Ground Pressure (kg/cm^2): INA

Automotive Performance:
Engine Type: 620-hp Diesel
Cruising Range (km): 450/650 with extra tanks
Speed (km/h):
Max Road: 45
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: N/A
Fording Depths (m): 1.4 Unprepared, 5.5 with snorkel

Radio: R-173, R-173P, R-124 intercom

Protection:
Armor, Turret Front (mm): 230
Applique Armor (mm): Bra armor (+100 on turret) and track skirts
Explosive Reactive Armor (mm): Available, replaces bra armor
Active Protective System: Russian Drozd APS will fit
Mineclearing Equipment: Roller-plow set, and plows
Self-Entrenching Blade: No
NBC Protection System: Nuclear radiation only
Smoke Equipment: Vehicle engine exhaust smoke system
2 x 4 Smoke grenade launchers

ARMAMENT
Main Armaments:
Caliber, Type, Name: 115-mm smoothbore gun, 2A20/Sheksna
Rate of Fire (rd/min): 3-5
Loader Type: Manual
Ready/Stowed Rounds: INA
Elevation (°): -5 to +18
Fire on Move: Yes (gun rounds only--ATGMs require a short halt)

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) machinegun PKT
Mount Type: Turret coax
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 800
Night: 800
Fire on Move: Yes
Rate of Fire (rd/min): 250 rpm practical, 800 cyclic, 2-10 rd bursts

ATGM Launcher:
Name: 2A20 gun
Launch Method: Gun-launched
Guidance: SACLOS, Infrared laser-beam rider
Command Link: Encoded laser-beam
Launcher Dismountable: No

FIRE CONTROL
FCS Name: Volna
Main Gun Stabilization: M1 Meteor 2-plane
Rangefinder: KTD-2 Laser
Infrared Searchlight: L-4
Sights w/Magnification:
Gunner:
Day: TShSM-41U, 3.5x and 7x
Field of View (°): 18 and 8
Acquisition Range (m): 4,000
Night: 1K13-1
Field of View (°): INA
Acquisition Range (m): 850-1,300, gun rounds only
Commander Fire Main Gun: No

VARIANTS
T-62M is one of a variety of T-62 variants. **T-62A:** added a 12.7-mm MG. **T-62M** adds protection, FCS and ATGM capability. T-62 variants with a V-46 T-72-type engine add **-1** to their designation.
T-62M1: Variant with Volna FCS but no missile launch capability.
T-62D: Variant with the Drozd APS vs ERA.
T-62MK: Command variant.
T-62MV: Version with ERA in place of the bra armor. The ERA includes Kontakt ERA and Kontakt-5 2nd-Generation ERA.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
115-mm APFSDS-T, BD/36-2
Maximum Aimed Range (m): 3,000
Max Effective Range (m):
Day: 2,000+ (est)
Night: 850-1,300
Armor Penetration (mm): 520 (RHA, 71° angle) at 1,000 m

115-mm APFSDS-T, BM-6 Russian
Maximum Aimed Range(m): 3,000
Max Effective Range (m):
Day: 1,500
Night: 850-1,300
Armor Penetration (mm): 237 (RHA) at 1,000 m

Russian Main Battle Tank T-62M

115-mm HEAT, BK-4 Maximum Aimed Range (m): 1,500 (est) Max Effective Range (m): Day: 1,200 Night: 850-1,200 Armor Penetration (mm): 495 (RHA) 115-mm Frag-HE-T, OF-27 Maximum Aimed Range (m): 4,000 Max Effective Range (m): Day: 1,500-2,000 Night: 850-1,300 Armor Penetration (mm): INA **Other Ammunition Types:** BM-3 APFSDS, BM-4 APFSDS, BK-4M HEAT, BK-15 HEAT, OF-11 Frag-HE, OF-18 Frag-HE	**Antitank Guided Missiles** Name: AT-10/Sheksna Warhead Type: Shaped charge (HEAT) Armor Penetration (mm): 650 Range (m): 4,000 (day only, see NOTES) Name: AT-10 Improved Warhead Type: Tandem shaped charge. Armor Penetration (mm): 700 behind ERA Range (m): 4,000 (day only, see NOTES)

NOTES

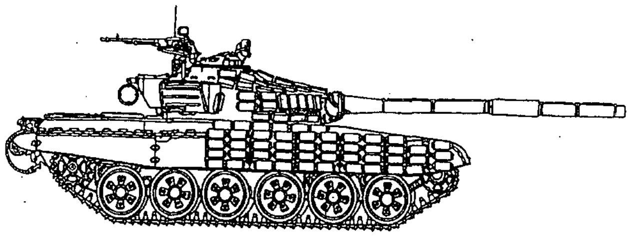
The 1K13 sight is both night sight and ATGM launcher sight; however, it cannot be used for both functions simultaneously.

Other improvements available include a hull bottom reinforced against mines, rubber track pads, and a thermal sleeve for the gun.

Optional sights and fire control systems include the Israeli El-Op Red Tiger and Matador FCS, Swedish NobelTech T-series sight, and German Atlas MOLF. The British Marconi Digital FCS, South African Tiger, and Belgian SABCA Titan offer upgraded function. One of the best is the Slovenian EFCS-3 integrated FCS.

A variety of thermal sights is available. They include the Russian Agava, French SAGEM-produced ALIS and Namut sight from Peleng. There are thermal sights available for installation which permit night launch of ATGMs.

Russian Main Battle Tank T-72B



T-72B w/Kontakt ERA

Weapons & Ammunition Types	Typical Combat Load
125-mm smoothbore gun	45
APFSDS-T	(mix est) 15
HEAT	3
Frag-HE	21
ATGM	6
7.62-mm coax MG	2,000
12.7-mm AA MG	300

SYSTEM
Alternative Designations: T-72S (export), SMT M1988
Date of Introduction: 1985
Proliferation: At least 2 countries
Description:
Crew: 3
Combat Weight (mt): 44.5
Chassis Length Overall (m): 6.91
Height Overall (m): 2.19
Width Overall (m): 3.58
Ground Pressure (kg/cm^2): 0.90

Automotive Performance:
Engine Type: 840-hp Diesel
Cruising Range (km): 500/ 900 with external tanks
Speed (km/h):
Max Road: 60
Max Off-Road: 45
Average Cross-Country: 35
Max Swim: N/A
Fording Depths (m): 1.2 Unprepared/5.0 with snorkel
Radio: R-173 and R-134

Protection:
Armor, Turret Front (mm): 520/950 against HEAT
Applique Armor (mm): Side of hull over track skirt, turret top
Explosive Reactive Armor (mm): Kontakt or Kontakt-5 ERA
Active Protective System: Arena available
Mineclearing Equipment: Roller-plow set, and plows available
Self-Entrenching Blade: Yes
NBC Protection System: Yes
Smoke Equipment: Smoke grenade launchers (8x 81-mm left side of turret), and 32 grenades. Vehicle engine exhaust smoke system.

ARMAMENT
Main Armament:
Caliber, Type, Name: 125-mm smoothbore gun 2A46M/ D-81TM
Rate of Fire (rd/min): 4-6/2 in manual mode
Loader Type: Autoloader (separate loading) and manual
Ready/Stowed Rounds: 22/23
Elevation (°): -6 to +14
Fire on Move: Yes, up to 25 km/h. Depending on the road and distance to the target, most crews may halt before firing.

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) Machinegun PKT
Mount Type: Turret coax
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,000
Night: 800
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical, 600 cyclic in 2-10 round bursts

Caliber, Type, Name: 12.7-mm (12.7x108) AA MG NSVT
Mount Type: Turret top
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,500/1,000 antiaircraft
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 200 practical, 600 cyclic in bursts

ATGM Launcher:
Name: 2A46M
Launch Method: Gun-launched
Guidance: SACLOS, Laser beam rider
Command Link: Encoded infrared laser beam
Launcher Dismountable: No

FIRE CONTROL
FCS Name: 1A40-1
Main Gun Stabilization: 2E42-2, 2-plane
Rangefinder: TPD-K1M laser rangefinder
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: TPD-K1, 8
Field of View (°): 9
Acquisition Range (m): 3,000 with LRF, 5,000 without
ATGM/Night: 1K13-495 5.6x (8x ATGM)
Field of View (°): 6, 40 min (5 ATGM)
Acquisition Range (m): INA
Commander Fire Main Gun: No

VARIANTS
T-72BK: Commander's variant with additional radios

T-72BM: Version with Kontakt-5 explosive reactive armor. This system is being fielded and is available for export.

T-72S/Shilden: Russian export T-72A upgraded to be comparable to the T-72BM standard. Although similar to the T-72BM, it may have less turret front protection. The early T-72S tank has Kontakt ERA, as shown above.

T-90: Successor to T-72BM. This tank has been tentatively approved for production and adoption as a standard tank, alongside the T-80U, for the Russian army. The T-90 uses the gun and 1G46 gunner sights from T-80U, a new engine, and thermal sights. Protective measures include Kontakt-5 ERA, laser warning receivers, and the SHTORA infrared ATGM jamming system.

Russian Main Battle Tank T-72B continued

MAIN ARMAMENT AMMUNITION	
Caliber, Type, Name: 125-mm APFSDS-T, BM-42M Maximum Aimed Range (m): 3,000 Max Effective Range (m): Day: 2,000-3,000 Night: 850-1,300 Armor Penetration (mm): 590-630 at 2,000 meters 125-mm Frag-HE-T, OF-26 Maximum Aimed Range (m): 5,000 Max Effective Range (m): Day: INA Night: 850-1,300 Armor Penetration (mm): INA 125-mm HEAT-MP, BK-29M Maximum Aimed Range (m): 3,000 Max Effective Range (m): Day: INA Night: 850-1300 Armor Penetration (mm): 650-750	125-mm HEAT, BK-27 Maximum Aimed Range (m): 3,000 Max Effective Range (m): Day: INA Night: 850-1,300 Armor Penetration (mm): 700-800 **Other Ammunition Types:** Giat 125G1 APFSDS-T, Russian BM-42 and BM-32 APFSDS-T. Note: The Russians may have a version of the BM-42M with a DU penetrator. **Antitank Guided Missiles:** Name: AT-11/SVIR Warhead Type: Shaped charge (HEAT) Armor Penetration (mm): 700 behind ERA/800 conventional Range (m): 4,000 Name: AT-11B/INVAR Warhead Type: Tandem Shaped charge (HEAT) Armor Penetration (mm): 800 behind ERA /870 conventional Range (m): 4,000

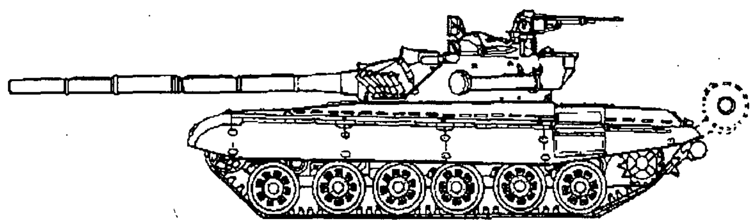
NOTES

The T-72B is the second main variant from the original Russian T-72 tank (after T-72A).

The 1K13-49 sight is both night sight and ATGM launch sight. However, it cannot be used for both functions simultaneously. A variety of thermal sights is available. They include the Russian Agava-2, French SAGEM-produced ALIS and Namut sight from Peleng. Thermal gunner night sights are available which permit night launch of ATGMs.

The more recent BK-27 HEAT round offers a triple-shaped charge warhead and increased penetration against conventional armors and ERA. The BK-29 round, with a hard penetrator in the nose is designed for use against reactive armor, and as an MP round has fragmentation effects. If the BK-29 HEAT-MP is used, it may substitute for Frag-HE (as with NATO countries) or complement Frag-HE. With three round natures (APFSDS-T, HEAT-MP, ATGMs) in the autoloader vs four, more antitank rounds would available for the higher rate of fire.

Polish/Czechoslovakian Main Battle Tank T-72M1



Weapons & Ammunition Types	Typical Combat Load
125-mm smoothbore gun	**44**
APFSDS-T	(mix est) 15
HEAT	7
Frag-HE	22
7.62-mm coax MG	**2,000**
12.7-mm AA MG	**300**

SYSTEM
Alternative Designations: Russian T-72A
Date of Introduction: 1975
Proliferation: At least 7 countries

Description:
Crew: 3
Combat Weight (mt): 41.5 (without ERA)
Chassis Length Overall (m): 6.91
Height Overall (m): 2.19
Width Overall (m): 3.59
Ground Pressure (kg/cm^2): 0.90

Automotive Performance:
Engine Type: 780-hp Diesel
Cruising Range (km): 460/700 with extra tanks
Speed (km/h):
Max Road: 60
Max Off-Road: 45
Average Cross-Country: 35
Max Swim: N/A
Fording Depths (m): 1.2 Unprepared/5.0 with snorkel

Radio: R-173M

Protection:
Armor, Turret Front (mm): 500/560 against HEAT
Applique Armor (mm): Side of hull over track skirt, turret top
Explosive Reactive Armor (mm): 1st or 2nd Gen ERA available
Active Protective System: Arena or Drozd available
Mineclearing Equipment: Roller-plow set, and plows available
Self-Entrenching Blade: Yes
NBC Protection System: Yes
Smoke Equipment: Smoke grenade launchers (6x 81-mm each side of turret), and 24 grenades. Vehicle engine exhaust smoke system.

ARMAMENT
Main Armaments:
Caliber, Type, Name: 125-mm smoothbore gun 2A46M/ D-81TM
Rate of Fire (rd/min): 4-6/2 in manual mode
Loader Type: Autoloader (separate loading) and manual
Ready/Stowed Rounds: 22/22 (22 in carousel)
Elevation (°): -6 to +14
Fire on Move: Yes, up to 25 km/h. Depending on the road and distance to the target, most crews may halt before firing.

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) Machinegun PKT
Mount Type: Turret coax
Maximum Aimed Range (m): 1,800
Max Effective Range (m):
Day: 1,000
Night: 800
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical, 600 cyclic in 2-10 round bursts

Caliber, Type, Name: 12.7-mm (12.7x108) AA MG NSVT
Mount Type: Turret top
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,500, 1,000 AA
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 200 practical, 600 cyclic in bursts

ATGM Launcher: N/A

FIRE CONTROL
FCS Name: INA
Main Gun Stabilization: 2E28M, 2-plane
Rangefinder: TPD-K1 laser rangefinder
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: TPD-K1 laser rangefinder sight, 8 x
Field of View (°): 9
Acquisition Range (m): 3,000 with LRF, 5000 without
Night: TPN-1-49, 5.5 x
Field of View (°): 6
Acquisition Range (m): 800
Commander Fire Main Gun: No

VARIANTS
T-72: Original Russian tank from which T-72 variants were derived.

T-72M: Original Polish and former-Czechoslovakian T-72-series tank from which Polish/Czechoslovakian T-72M1 was derived.
T-72M differs from T-72 in replacing the right-side coincident rangefinder with a centerline-mounted TPDK-1 LRF.

T-72A: The Russian variant differs from T-72 with the TPDK-1 LRF, added sideskirts, additional armor on the turret front and top, smoke grenade launchers, internal changes, and a slight weight increase. The Russian export version and Polish/Czechoslovakian counterparts are called **T-72M1**. Versions with Kontact ERA are known as **T-72AV /T-72 M1V**. Please note that some countries have inventories of T-72, T-72M and T-72M1, with different versions of each variant. Also, many variants were upgraded or modified. Some T-72M1s do not have smoke grenade launchers or track skirts. Some T-72s/T-72Ms have smoke grenade launchers. More reliable discriminators are armor and rangefinder/FCS.

Polish/Czechoslovakian Main Battle Tank T-72M1 continued

T-72AK/7T-2M1K: Commander's variant with additional radios

T-72AM/Banan: Ukrainian T-72A upgrade with ERA, a new engine, and additional smoke grenade launchers. The **T-72AG** upgrade has a 1200-hp engine, Shtora-1 ATGM jammer, and 1G46 (T-80U) FCS with thermal night sights.

T-72M1M: T-72M1 variant upgraded to T-72B standard.

T-72M2/Moderna. Slovakian T-72M upgrade with new engine and fire control, SFIM thermal sight, laser warning receiver, ERA, and 2 x 20-mm AA guns on turret

T-72M4CZ: Czech variant with TURMS FCS with thermal sight, new engine, increased protection ERA, and 48t weight. **T72M3CZ** ia a less radical upgrade-- for instance existing engine is modified.

T-72MP: Ukrainian upgrade with a 1,000-hp engine, added armor, Shtora-1, and SAGEM FCS and thermal sights.

T-72S/Shilden: Russian export T-72A upgraded to T-72B standard.

M-84: Former Yugoslavian tank upgraded to T-72M1 standard, but with indigenous sights. With an upgraded engine, the tank is **M-84A**. A Croatian improved version of M-84 is **M84A4/Sniper**, with improved fire control and thermal night sights. A Slovenian upgrade uses the state-of-the-art and the well-marketed EFCS-3 FCS.

PT-91/Twardy: Polish upgrade tank with ERA, laser warning receiver, smoke grenade launchers, and Tiger fire control system. Sights include a thermal gunner night sight.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:

125-mm APFSDS-T, BM-42M
- Maximum Aimed Range (m): 3,000
- Max Effective Range (m):
  - Day: 2,000-3,000
  - Night: 850-1,300
- Armor Penetration (mm): 590-630 at 2,000 meters

125-mm Frag-HE-T, OF-26
- Maximum Aimed Range (m): 5,000
- Max Effective Range (m):
  - Day: INA
  - Night: 850-1,300
- Armor Penetration (mm): INA

125-mm HEAT-MP, BK-29M
- Maximum Aimed Range (m): 3,000
- Max Effective Range (m):
  - Day: INA
  - Night: 850-1300
- Armor Penetration (mm): 650-750

125-mm HEAT, BK-27
- Maximum Aimed Range (m): 3,000
- Max Effective Range (m):
  - Day: INA
  - Night: 850-1,300
- Armor Penetration (mm): 700-800

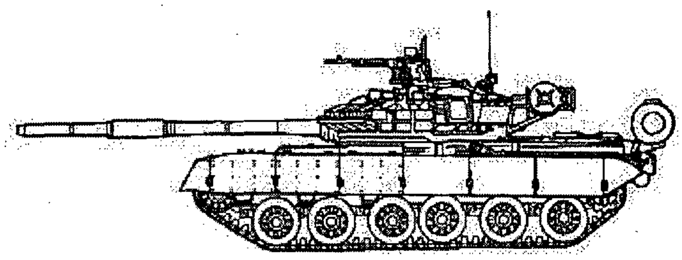
Other Ammunition Types: Giat 125G1 APFSDS-T, Russian BM-42 and BM-32 APFSDS-T. Note: The Russians may have a version of the BM-42M with a DU penetrator.

NOTES

A variety of thermal sights is available. They include the Russian Agava-2, French SAGEM-produced ALIS and Namut sight from Peleng.

The more recent BK-27 HEAT round offers a triple-shaped charge warhead and increased penetration against conventional armors and ERA. The BK-29 round, with a hard penetrator in the nose is designed for use against reactive armor, and as an MP round has fragmentation effects. If the BK-29 HEAT-MP is used, it may substitute for Frag-HE (as with NATO countries) or complement Frag-HE. With three round natures (APFSDS-T, HEAT-MP, ATGMs) in the autoloader vs four, more antitank rounds would available for the higher rate of fire.

Russian Main Battle Tank T-80B



Weapons & Ammunition Types	Typical Combat Load
125-mm smoothbore gun	**45**
APFSDS-T	(mix est) 15
HEAT	3
Frag-HE	21
ATGM	6
7.62-mm coax MG	**1,250**
12.7-mm NSVT AA MG	**500**

SYSTEM
Alternative Designations: See NOTES
Date of Introduction: 1978
Proliferation: At least 1 country
Description:
Crew: 3
Combat Weight (mt): 44.5
Chassis Length Overall (m): 6.98
Height Overall (m): 2.22
Width Overall (m): 3.58
Ground Pressure (kg/cm^2): 0.87

Automotive Performance:
Engine Type: 1,000-hp or 1,100-hpGas turbine (multifuel),
Cruising Range (km): 370/ 500 with extra tanks
Speed (km/h):
Max Road: 70
Max Off-Road: 48
Average Cross-Country: 40
Max Swim: N/A
Fording Depths (m): 1.8 Unprepared, 5.0 w/snorkel, 12.0 with BROD-M system

Radio: R-173, R-174 intercom

Protection:
Armor, Turret Front (mm): Defeat 120-mm rounds (triple layer)
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): 1st Generation ERA available
Active Protective System: Available
Mineclearing Equipment: Mine rollers and plows available
Self-Entrenching Blade: Yes
NBC Protection System: Yes
Smoke Equipment: Smoke grenade launchers (4x 81-mm each side of turret), and 24 grenades. Vehicle engine exhaust smoke system

ARMAMENT
Main Armaments:
Caliber, Type, Name: 125-mm smoothbore gun 2A46-2
Rate of Fire (rd/min): 6-8 (lower in manual mode)
Loader Type: KORZINA separate-loading autoloader and manual
Ready/Stowed Rounds: 28 in carousel/17 rounds stowed but readily available for manual loading
Elevation (°): -7 to +20
Fire on Move: Yes (30 km/h gun rounds/low speed or stop ATGMs)

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) Machinegun PKT
Mount Type: Turret coax
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,000
Night: 850-1,300
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, 2-10 round bursts

Caliber, Type, Name: 12.7-mm (12.7x108) AA MG NSVT
Mount Type: Turret top
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,500 ground/1,600 for air targets (APDS)
Night: 800-1,300
Fire on Move: Yes
Rate of Fire (rd/min): 210 practical/ 800 air targets in bursts

ATGM Launcher:
Name: 2A46-2 tank gun
Launch Method: Gun-launched
Guidance: SACLOS
Command Link: Encoded radio frequency
Launcher Dismountable: No

FIRE CONTROL
FCS Name: FCS 1A33
Main Gun Stabilization: 2E26M 2-plane
Rangefinder: Laser
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: 1G42
Field of View (°): INA
Acquisition Range (m): 5,000
Night: 1-4A
Field of View (°): INA
Acquisition Range (m): 800-1,300 (est)
Commander Fire Main Gun: No

VARIANTS
T-80BV: Variant noted in the above line drawing has ERA mounted. This variant is more likely for encounter by US forces.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
125-mm APFSDS-T, BM-42M
Maximum Aimed Range (m): 3,000-4,000
Max Effective Range (m):
Day: 2,000-3,000
Night: 850-1,300
Armor Penetration (mm): 590-630 at 2,000 meters

Russian Main Battle Tank T-80B continued

125-mm Frag-HE-T, OF-26 Maximum Aimed Range (m): 5,000 Max Effective Range (m): Day: INA Night: 850-1,300 Armor Penetration (mm): INA 125-mm HEAT-MP, BK-29M Maximum Aimed Range (m): 4,000 Max Effective Range (m): Day: 2,000-3,000 Night: 850-1300 Armor Penetration (mm): 650-750 125-mm HEAT, BK-27 Maximum Aimed Range (m): 4,000 Max Effective Range (m): Day: 2,000-3,000 Night: 850-1,300 Armor Penetration (mm): 700-800	**Other Ammunition Types:** Giat 125G1 APFSDS-T, Russian BM-42 and BM-32 APFSDS-T. Note: The Russians may have a version of the BM-42M with a DU penetrator. **Antitank Guided Missile:** Name: AT-8/SONGSTER Warhead Type: Shaped charge (HEAT) Armor Penetration (mm): 700 (RHA) conventional Range (m): 4,000

NOTES

The T-80B and -BV variants are often misidentified as T-80. They are visibly different and bear other distinctions, such as T-80B/-BV capability for launching AT-8/ Songster ATGM.

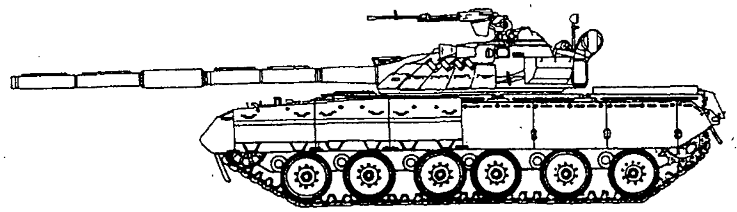
The night sight cannot be used to launch the ATGM. The daysight can be used at night for launching ATGMs if the target is illuminated.
A variety of thermal sights is available. They include the Russian Agava-2, French SAGEM-produced ALIS and Namut sight from Peleng. There are thermal sights available for installation which permit night launch of ATGMs.

The 12.7-mm MG NSVT has both remote electronically operated sight PZU-5 and gun-mounted K10-T reflex sight.

The more recent BK-27 HEAT round offers a triple-shaped charge warhead and increased penetration against conventional armors and ERA. The BK-29 round, with a hard penetrator in the nose is designed for use against reactive armor, and as an MP round has fragmentation effects. If the BK-29 HEAT-MP is used, it may substitute for Frag-HE (as with NATO countries) or complement Frag-HE. With three round natures (APFSDS-T, HEAT-MP, ATGMs) in the autoloader vs four, more antitank rounds would available for the higher rate of fire.

The ATGM may be launched while moving slowly (NFI). The AT-8 can be auto-loaded with the two halves mated during ramming; but the stub charge is manually loaded.

Russian Main Battle Tank T-80U



Weapons & Ammunition Types	Typical Combat Load
125-mm smoothbore gun	**45**
APFSDS-T	(mix est) 15
HEAT	3
Frag-HE	21
ATGM	6
7.62-mm coax MG	**1,250**
12.7-mm NSVT AA MG	**500**

SYSTEM
Alternative Designations: SMT (Soviet Medium Tank) M1989
Date of Introduction: 1987
Proliferation: At least 3 countries
Description:
Crew: 3
Combat Weight (mt): 46.0
Chassis Length Overall (m): 7.01
Height Overall (m): 2.20
Width Overall (m): 3.60
Ground Pressure (kg/cm^2): 0.92

Automotive Performance:
Engine Type: 1250-hp Gas turbine (multi-fuel), diesel on T-80UD
Cruising Range (km): 335 km/600 km with extra tanks
Speed (km/h):
Max Road: 70
Max Off-Road: 48
Average Cross-Country: 40
Max Swim: N/A
Fording Depths (m): 1.8 Unprepared, 5.0 w/snorkel, 12.0 with BROD-M system

Radio: R-173, R-174 intercom

Protection:
Armor, Turret Front (mm): Against 120-mm ammunition
Applique Armor (mm): Side of hull, over track skirt
Explosive Reactive Armor (mm): Kontakt-5 2nd Generation ERA
Active Protective System: ARENA is available
Mineclearing Equipment: Roller-plow set and plows available
Self-Entrenching Blade: Yes
NBC Protection System: Yes
Smoke Equipment: Smoke grenade launchers (4x 81-mm each side of turret), and 24 grenades. Vehicle engine exhaust smoke system.

ARMAMENT
Main Armaments:
Caliber, Type, Name: 125-mm smoothbore gun 2A46M-1
Rate of Fire (rd/min): 7-8 (lower in manual mode)
Loader Type: KORZINA separate-loading autoloader, and manual
Ready/Stowed Rounds: 28 in carousel/17 stowed (manual loaded)
Elevation (°): -4 to +18
Fire on Move: Yes (gun rounds and ATGMs)

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) Machinegun PKT
Mount Type: Turret coaxial
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 800
Night: 800
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical / 650 cyclic, 2-10 round bursts

Caliber, Type, Name: 12.7-mm (12.7x108) AA MG NSVT
Mount Type: Turret top
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,500
Night: 800-1,300
Fire on Move: Yes
Rate of Fire (rd/min): 210 practical/ 800 air targets in bursts

ATGM Launcher:
Name: 2A46M-1 tank gun
Launch Method: Gun-launched
Guidance: SACLOS, Laser-beam rider
Command Link: Encoded infrared laser-beam
Launcher Dismountable: No

FIRE CONTROL
FCS Name: FCS 1A42
Main Gun Stabilization: 2342, 2-plane
Rangefinder: Laser
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: 1G46/PERFECT, 3.6/12x
Field of View (°): INA
Acquisition Range (m): 5,000 (70%P-hit for ATGM)
Night: AGAVA-2
Field of View (°): INA
Acquisition Range (m): 2,600 (gun rounds only)
Commander Fire Main Gun: Yes

VARIANTS
T-80UD: Version produced in the Ukraine with a 1000-hp diesel engine instead of the turbine engine, and 1st generation ERA.

T-80UK: Command version with R-163-50K and R-163-U radios, TNA-4 land navigation system, and an electronic fuze-setting device that permits use of Ainet Shrapnel Round. The AGAVA thermal sight provides a 2,600-meter night acquisition range.

T-84: Recent Ukrainian upgrade of T-80UD with a welded turret, a French ALIS thermal sight, a more powerful engine, optional use of ARENA active protection system (APS) and SHTORA-1 active IR ATGM jammer system. Prototypes have been demonstrated, and the tank is available for export.

Russian Main Battle Tank T-80U continued

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
125-mm APFSDS-T, BM-42M
Maximum Aimed Range (m): 3,000-4,000
Max Effective Range (m):
Day: 2,000-3,000
Night: 800-1,300
Armor Penetration (mm): 590-630 at 2,000 meters

125-mm HE-Shapnel Focused-fragmentation, Ainet
Maximum Aimed Range (m): 5,000
Max Effective Range (m):
Day: 4,000
Night: 800-1,300
Tactical AA Range: 4,000-5,000
Armor Penetration (mm): INA

125-mm Frag-HE-T, OF-26
Maximum Aimed Range (m): 5,000
Max Effective Range (m):
Day: INA
Night: 800-1,300
Armor Penetration (mm): INA

125-mm HEAT-MP, BK-29M
Maximum Aimed Range (m): 4,000
Max Effective Range (m):
Day: INA
Night: 800-1300
Armor Penetration (mm): 650-750

125-mm HEAT, BK-27
Maximum Aimed Range (m): 4,000
Max Effective Range (m):
Day: INA
Night: 800-1,300
Armor Penetration (mm): 700-800

Other Ammunition Types: Giat 125G1 APFSDS-T, Russian BM-42 and BM-32 APFSDS-T. Note: The Russians may have a version of the BM-42M with a DU penetrator.

Antitank Guided Missiles:
Name: AT-11/SVIR
Warhead Type: Shaped charge (HEAT)
Armor Penetration (mm): 700 (RHA) behind ERA/800 conventional
Range (m): 5,000

Name: AT-11B/INVAR
Warhead Type: Tandem shaped charge
Armor Penetration (mm): 800 (RHA) behind ERA /870 conventional
Range (m): 5,000

NOTES
Line drawing is a T-80UD.

GTA-18A Auxiliary Power Unit is used when the engine is off.

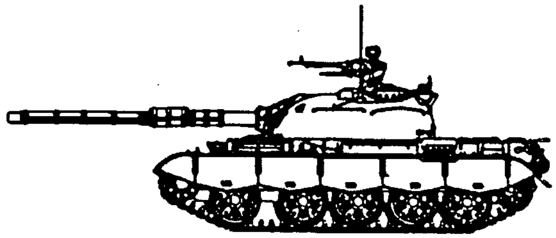
The BK-29 round, with a hard penetrator in the nose is designed for use against reactive armor, and as an MP round has fragmentation effects. The more recent BK-27 HEAT round offers a triple-shaped charge warhead and 50 mm more penetration.

The electronic round fuzing system for Ainet rounds is available for other tanks. This round uses technology similar to that for French Oerlikon's AHEAD rouns. The round is specially designed to defeat targets by firing fragmentation patterns forward and radially, based on computer calculated settings from the laser range-finder and other inputs. Targets are helicopters and dug in or defilade priority ground threats, such as ATGM positions. Rate of fire is 4 rd/min.

The 12.7-mm MG NSVT has both remote electronically operated sight PZU-5 and gun-mounted K10-T reflex sight.

The original night sight is the II Buran-PA (800-1300 meters range). The sight cannot be used to launch the ATGM. The daysight can be used at night for launching ATGMs if the target is illuminated. A variety of thermal sights is available. They include the Russian Agava-2, French SAGEM-produced ALIS and Namut sight from Peleng. There are thermal sights available for installation which permit night launch of ATGMs.

Chinese Main Battle Tank Type 59-II



Weapons & Ammunition Types	Typical Combat Load
105-mm rifled gun L7	**34**
New CH APFSDS-T	12
M456 HEAT	6
L35 HESH	16
7.62-mm coax MG	**2,000**
7.62-mm bow MG	**1,000**
12.7-mm AA MG	**500**

SYSTEM
Alternative Designations: WZ 120B
Date of Introduction: 1951
Proliferation: At least 2 countries
Description:
Crew: 4
Combat Weight (mt): 36.5-37.0
Chassis Length Overall (m): 6.04
Height Overall (m): 2.59
Width Overall (m): 3.30
Ground Pressure (kg/cm^2): 0.8

Automotive Performance:
Engine Type: 520-hp Diesel
Cruising Range (km): 440/600 with external tanks
Speed (km/h):
Max Road: 50
Max Off-Road: 25
Average Cross-Country: INA
Max Swim: N/A
Fording Depths (m): 1.4 Unprepared, 5.5 with snorkel

Radio: INA

Protection:
Armor, Turret Front (mm): 203
Applique Armor (mm): Track skirts are fitted to some tanks
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: Mine plows and roller-plows available
Self-Entrenching Blade: N/A
NBC Protection System: N/A
Smoke Equipment: 8 x 81-mm smoke grenade launchers
Vehicle engine exhaust smoke system

ARMAMENT
Main Armaments:
Caliber, Type, Name: 105-mm rifled gun, similar to L7
Rate of Fire (rd/min): 6-10
Loader Type: Manual
Ready/Stowed Rounds: INA
Elevation (°): -5/+18
Fire on Move: Yes

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) Machine gun Type 59T
Mount Type: Turret coax
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,000
Night: 800
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical, 600 cyclic in 2-10 round bursts

Caliber, Type, Name: 7.62-mm (7.62x 54R) Machine gun Type 59T
Mount Type: Bow ball mount
Maximum Aimed Range (m): 1,000
Max Effective Range (m):
Day: 1,000
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical, 600 cyclic in 2-10 round bursts

Caliber, Type, Name: 12.7-mm (12.7x108) AA MG Type 54
Mount Type: Turret cupola
Maximum Aimed Range (m): 2.000
Max Effective Range (m):
Day: 1,500 ground/1,600 for air targets (APDS)
Night: N/A, II sights available
Fire on Move: Yes
Rate of Fire (rd/min): 80-100 practical, 600 air targets 2-10 rd bursts

FIRE CONTROL
FCS Name: UI light spot fire control system
Main Gun Stabilization: 2-plane
Rangefinder: LRF
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: INA
Field of View (°): INA
Acquisition Range (m): INA
Night: Type DC 1024/00 II sights, x7
Field of View (°): 6
Acquisition Range (m): 1,000
Commander Fire Main Gun: No

VARIANTS:
Type 59: Original model is a copy of the Former Soviet T-54 MBT and has a 100-mm main gun.

T-72Z/ Safir 74: Iranian variant which constitutes state of the art for upgraded 50s-generation former Warsaw Pact tanks. This tank has a 780-hp diesel engine, track skirts, and smoke grenade launchers. An Iranian ERA package will fit T-72Z. Armament includes an M68 105-mm rifled gun, 7.62-mm Type 59T (PKT) MG, and a 12.7-mm Type 59 (DShKM) MG. The cannon can launch AT-10/ Bastion ATGMs (to 4000 meters) and fire a broad range of NATO 105-mm ammunition. Fire control includes the robust Slovenian EFCS-3-55 fire control system with stabilization, a laser rangefinder, and a ballistic computer. The FCS includes a commander's independent viewer and target designation system, and II gunner night sights.

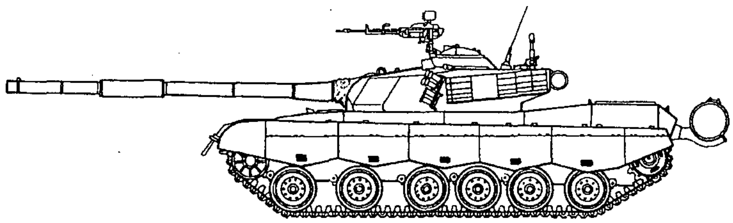
Chinese Main Battle Tank Type 59-II continued

MAIN ARMAMENT AMMUNITION **Caliber, Type, Name:** 105-mm APFSDS, H6/62 Maximum Aimed Range (m): 3,000 Max Effective Range (m): Day: 2,000-3,000 (est) Night: 800-1,300 Armor Penetration (mm): INA 105-mm APFSDS, UI (New Chinese) Maximum Aimed Range (m): 3,000 Max Effective Range (m): Day: 2,000-3,000 (est) Night: 800-1,300 Armor Penetration (mm): 460 at 2,000 m	105-mm HEAT, M456 (multinational) Maximum Aimed Range (m): 3,000 Max Effective Range (m): Day: 1,500-2,500 (est) Night: 800-1,300 Armor Penetration (mm): 432, NATO single heavy target 105-mm HESH, L35 (UK) Maximum Aimed Range (m): 5,000 Max Effective Range (m): Day: 2,000-3,000 (est) Night: 800-1,300 Armor Penetration (mm): NATO single heavy target **Other Ammunition Types:** Chinese Type 83/ UK L64/ US M735 APFSDS, UK L52 APDS, multinational M393 HEP-T, French OE 105-F1 HE, L39 Smoke, cannister

NOTES

GEC-Marconi Centaur fire control system is available. British Barr and Stroud thermal based FCS can be fitted.

Chinese Main Battle Tank Type 85-IIM



Weapons & Ammunition Types	Typical Combat Load
125-mm smoothbore gun	**42**
APFSDS-T	(mix est) 15
HEAT	6
Frag-HE	21
7.62-mm coax MG	**2,000**
12.7-mm cupola AAMG	**500**

SYSTEM
Alternative Designations: INA
Date of Introduction: 1991
Proliferation: At least 2 countries
Description:
Crew: 3
Combat Weight (mt): 41.0
Chassis Length Overall (m): 10.28
Height Overall (m): 2.30
Width Overall (m): 3.450
Ground Pressure (kg/cm^2): 0.771

Automotive Performance:
Engine Type: 730-hp Diesel
Cruising Range (km): 700/900 with external tanks
Speed (km/h):
Max Road: 57
Max Off-Road: 45
Average Cross-Country: 35
Max Swim: N/A
Fording Depths (m): 1.4 Unprepared, 2.4 with snorkel

Radio: INA

Protection:
Armor, Turret Front (mm): INA
Applique Armor (mm): Track skirts. Composite panels available.
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
Mineclearing Equipment: Mine plows and roller-plow set
Self-Entrenching Blade: N/A
NBC Protection System: Yes
Smoke Equipment: 12x 81-mm smoke grenade launchers
Vehicle engine exhaust smoke system

ARMAMENT
Main Armament:
Caliber, Type, Name: 125-mm smoothbore gun 2A46M/ D-81TM
Rate of Fire (rd/min): 4-6/2 in manual mode
Loader Type: Autoloader (separate loading) and manual
Ready/Stowed Rounds: 22/23 (22 in carousel)
Elevation (°): -6 to +14
Fire on Move: Yes, up to 25 km/h. Depending on the road and distance to the target, most crews may halt before firing.

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x 54R) Machine gun Type 59
Mount Type: Turret coax
Maximum Aimed Range (m): 1,800
Max Effective Range (m):
Day: 1,000
Night: 800
Fire on Move: Yes
Rate of Fire (rd/min): 250 practical, 600 cyclic, 2-10 rd bursts

Caliber, Type, Name: 12.7-mm (12.7x108) AA MG Type 54
Mount Type: Cupola
Maximum Aimed Range (m): 2,000
Max Effective Range (m):
Day: 1,500 ground/1600 for air targets (APDS)
Night: N/A
Fire on Move: Yes
Rate of Fire (rd/min): 80-100 practical, 600 air targets, 2-10 rd bursts

FIRE CONTROL
FCS Name: ISFCS-212 (Image-Stabilized Fire Control System)
Main Gun Stabilization: 2-plane
Rangefinder: LRF
Infrared Searchlight: Yes
Sights w/Magnification:
Gunner:
Day: UI stabilized gunner sight
Field of View (°): INA
Acquisition Range (m): INA
Night: 2nd Generation II sights
Field of View (°): INA
Acquisition Range (m): INA
Commander Fire Main Gun: No

VARIANTS
Type 85-IIAP: Variant assembled from Type 59s and Type 69-IIs and upgrade kits, or from licensed production in Pakistan.

Type 85-III: Upgraded variant with 1,000-hp engine and composite armor panels. Variant is in prototype stage.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
125-mm APFSDS-T, BM-42M
Maximum Aimed Range (m): 3,000
Max Effective Range (m):
Day: 2,000-3,000
Night: 850-1,300
Armor Penetration (mm): 590-630 at 2,000 meters

125-mm Frag-HE-T, OF-26
Maximum Aimed Range (m): 5,000
Max Effective Range (m):
Day: INA
Night: 850-1,300
Armor Penetration (mm): INA

Chinese Main Battle Tank Type 85-IIM continued

125-mm HEAT-MP, BK-29M Maximum Aimed Range (m): 3,000 Max Effective Range (m): Day: INA Night: 850-1300 Armor Penetration (mm): 650-750 125-mm HEAT, BK-27 Maximum Aimed Range (m): 3,000 Max Effective Range (m): Day: INA Night: 850-1,300 Armor Penetration (mm): 700-800	**Other Ammunition Types:** Giat 125G1 APFSDS-T, Russian BM-42 and BM-32 APFSDS-T. Note: The Russians may have a version of the BM-42M with a DU penetrator.

NOTES

GEC-Marconi Centaur fire control system is available. British Barr and Stroud thermal based FCS can be fitted.

The more recent BK-27 HEAT round offers a triple-shaped charge warhead and increased penetration against conventional armors and ERA. The BK-29 round, with a hard penetrator in the nose is designed for use against reactive armor, and as an MP round has fragmentation effects. If the BK-29 HEAT-MP is used, it may substitute for Frag-HE (as with NATO countries) or complement Frag-HE. With three round natures (APFSDS-T, HEAT-MP, ATGMs) in the autoloader vs four, more antitank rounds would available for the higher rate of fire.

Chapter 5
Antitank

As armored combat vehicles have ascended in importance on the battlefield, so have the systems designed to stop those vehicles. The umbrella term ***antitank*** originally denoted systems specifically designed to destroy tanks. But today it is also more broadly constructed. Modern combat is combined arms combat. Mechanized forces include other armored combat vehicles, such as armored reconnaissance vehicles, infantry fighting vehicles, armored personnel carriers, etc. Tanks cannot survive or achieve their tactical objectives without support from other armored systems. The more recent term ***antiarmor*** may supplant the current term; because antitank weapons which cannot penetrate tank armor can still be a formidable threat if they can defeat or damage more lightly armored fighting vehicles. With upgrades and innovative tactics even older, seemingly obsolete, weapons can be used as OPFOR antiarmor weapons.

Antitank weapons can include guns of various sizes, antitank guided missile launcher systems, rocket and grenade launchers, mines and their delivery systems, and other obstacle systems. The rocket and grenade launchers are described in Chapter 1, Infantry Weapons. Mines and other obstacle systems are noted at Chapter 8, Engineer Systems. Because the OPFOR place a high priority on stopping and destroying armored combat vehicles, they will use all other available assets which can doctrinally support the effort. These include fixed and rotary-wing aircraft, artillery, NBC assets, etc. A number of recent systems have been fielded seemingly for other roles, but available for use as antitank weapons: light tanks, heavy armored reconnaissance vehicles with guns of 60 millimeters or more, assault vehicles, fire support vehicles, and artillery/mortar-type combination guns, such as Russian 120-mm 2S9, 2S23, and 2S31. Many OPFOR countries will employ antitank weapons for roles other than antitank, including AT guns against personnel and soft targets, and ATGMs against personnel and rotary-wing aircraft.

Antitank guns include towed guns and self-propelled antitank guns (also known as tank destroyers). A number of guns were designed as field guns, with multi-role capability as both artillery and antitank guns. The modern focus on maneuver warfare has brought a slight decline in development of uniquely antitank guns. Thus, the 85-mm D-44 gun, which can be used as artillery, is effective for use in an antitank role. Although recent systems have been developed, the number fielded has not kept pace with production of armored combat vehicles. Nevertheless, their effectiveness and selected armies' continued reliance on linear positional battles and protracted defenses have kept a large number of these systems in inventories. Based on numbers fielded and likelihood of their threat to US forces, only towed antitank guns were included.

A number of upgrades are available. These include night sights, such as passive image intensifier sights and thermal sights for the Russian 100-mm MT-12. This is a robust antitank weapon, with a high rate of fire and rapid mobility. Note the Russian innovation in the MT-12R, an AT gun with a radar-directed all-weather fire control system. Improved ammunition is critical for continued effectiveness of antitank weapons. The MT-12 and its variants can fire a variety of modern ammunition, including the Russian gun-launched ATGM, Kastet.

The ***antitank guided missile*** (ATGM) is the singular greatest threat to tanks today. These systems are distinguished from other antitank weapons in that they are guided to the target. Most employ SACLOS guidance (see Glossary). An operator holds crosshairs on the target, and the missile tracker directs the missile to that point. There is a wide variety of countermeasures (such as smoke and counterfire, due to long flight time and operator vulnerability) for use against ATGMs. Thus, a 90% probability of hit is a technical figure, and does not mean a 90% probability of success. On the other hand, there is a variety of counter-countermeasures which the ATGMs, launchers, and operators can use to increase the chance for success. Tactics, techniques and procedures within the antitank arena are critical to mission success.

As armor protection levels and antitank weapon lethality levels continue to rise, armor protection for many modern tanks has outpaced most AT weapons. However, ATGMs have been able to increase their size, range, and warhead configurations to threaten even the heaviest tanks. Among notable trends in ATGMs is the worldwide proliferation and variety of manportable and portable ***antitank guided missile launchers***. These include shoulder-launched, short-range systems, such as the French Eryx, and a variety of copies of former Soviet systems, such as the AT-3/Malyutka ("Suitcase SAGGER). Another notable trend is in development of upgrade ATGMs, with increased lethality. The most common type of lethality upgrade is addition of a nose precursor or tandem warhead. A more recent lethality upgrade has been the use of warheads that permit the "fly-over, shoot-down" mode. These missiles can over-fly a vehicle behind a hill, and fire an explosively-formed penetrator (EFP, in the shape of a cannon kinetic-energy penetrator round) downward through the relatively soft top of armored vehicles. Other improvements include improved guidance and resistance to countermeasures, reduced smoke and noise signature, and increased range. A fairly common trend has been addition of night sights, including thermal sights for the launcher. As the missiles and launchers have been improved, weight loads have increased. Most of the so-called portable launchers (AT-4 launcher, TOW, and HOT) have outgrown the portability weight limit, and must be carried in vehicles and only dismounted short distances from the carriers.

Although there are unique ***ATGM launcher vehicles*** with unique ATGMs, most numerous launcher vehicles are military and commercial vehicles adapted with pintel mounts for portable ground launchers, with ATGMs manually loaded and launched. Configurations of those vehicles consist of simply pairing of vehicle and launcher, and can be executed with equipment at hand; therefore, they were not described in this guide.. The number of fielded ATGM launcher vehicles specially designed for the mission numbers no more than a few dozen systems. They constitute a high level threat to vehicles and rotary-winged aircraft in the US Army.

Systems selected for this chapter are the more common threat systems, or represent the spectrum of antitank systems which can threaten US Army forces in the world today.

Questions and comments on data listed in this chapter should be addressed to:

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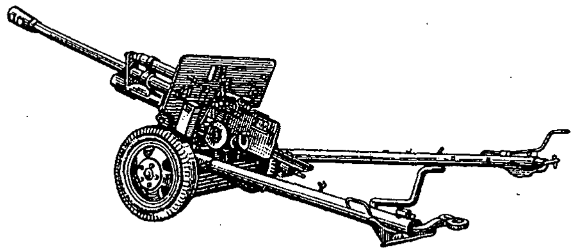
Russian 76-mm Towed Antitank Gun ZIS-3



Weapons & Ammunition Types	Typical Combat Load
76-mm rifled gun	INA
HVAP-T	
HEAT	
APC-T	
Frag-HE	

SYSTEM
Alternative Designations: M1942
Date of Introduction: INA
Proliferation: At least 14 countries

Description:
Crew: 5-7
Combat Weight (mt): 1.12
Length Overall, Travel (m): 6.10
Height Overall, Travel (m): 1.3
Width Overall, Travel (m): 1.4

Mobility:
Mount: Two-wheeled carriage with twin trails
Prime mover: AT-P tractor, light trucks
Towed Speed (km/h):
Max Road: INA
Max Off-Road: INA
Average Cross-Country: INA
Fording Depth (m): N/A
Emplace Time (min): INA
Displace Time (min): INA
Radio: N/A
Protection: Gun shield

ARMAMENT
Main Armaments:
Caliber, Type, Name: 76-mm rifled gun
Rate of Fire (rd/min): 8-10 normal / 15-20 burst indirect fire
Loader Type: Manual
Ready/Stowed Rounds: N/A
Elevation (°): -5/+37
Fire on Move: No

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Sights w/Magnification:
Gunner:
Day: INA
Field of View (°): INA
Acquisition Range (m): INA
Night: INA
Field of View (°): INA
Acquisition Range (m): INA

VARIANTS N/A

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
76-mm HVAP-T, BR-354P
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 1,000
Night: INA
Armor Penetration (mm): 58 at 1,000 meters
92 at 500 meters

76-mm APC-T , BR-350B
Maximum Aimed Range (m): INA
Max Effective Range (m): INA
Day: 1,000
Night: INA
Armor Penetration (mm): 61 at 1,000 meters

76-mm HEAT, BK-354M
Maximum Aimed Range (m): 1,000
Max Effective Range (m): INA
Day: 500
Night: INA
Armor Penetration (mm): 120 (RHA)

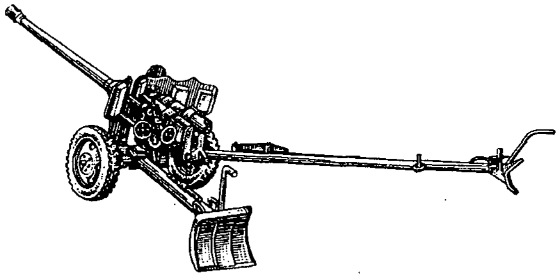
76-mm Frag-HE, OF-350A
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: INA
Night: INA
Armor Penetration (mm): INA

76-mm Frag-HE
Maximum Aimed Range (m): INA
Max Effective Range (m):
Day: 1,500
Night: INA
Armor Penetration (mm): INA

Other Ammunition Types: API-T BZR-350B, Smoke (WP)

NOTES
Although the ZIS-3 is categorized as an antitank gun, some OPFOR forces will employ it for general support, especially against light targets. Typical combat load is based on the prime mover; and a wide variety of systems can be used as prime movers.

Russian 85-mm Towed Gun D-44



Weapons & Ammunition Types	Typical Combat Load
85-mm rifled gun	**21**
HVAP-T	(est) 3
HEAT-FS	3
AP HE	3
Frag-HE	9
Smoke	3

SYSTEM
Alternative Designations: M1945
Date of Introduction: 1944
Proliferation: At least 16 countries
Description:
Crew: 8
Combat Weight (mt): 3.1
Length Overall, Travel (m): 8.34
Height Overall, Travel (m): 1.42
Width Overall, Travel (m): 1.73

Mobility:
Mount: Two-wheeled carriage with twin trails and coaster wheel
Prime mover: AT-P tractor, light trucks
Towed Speed (km/h):
Max Road: 60
Max Off-Road: 35
Average Cross-Country: INA
Fording Depth (m): INA
Emplace Time (min): 2
Displace Time (min): 2
Radio: N/A
Protection: Gun shield

ARMAMENT
Main Armaments:
Caliber, Type, Name: 85-mm rifled gun
Rate of Fire (rd/min): 8 normal / 15 burst Indirect Fire
Loader Type: Manual
Ready/Stowed Rounds: 0 / 140 on prime mover
Elevation (°): -7/+35
Fire on Move: No

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Sights w/Magnification:
Gunner:
Day: OP-2-7 Direct Fire, 5.5x / PG-1M Indirect Fire
Field of View (°): INA
Acquisition Range (m): 1,500
Night: INA
Field of View (°): INA
Acquisition Range (m): INA

VARIANTS
D-44-N: Variant with II night sight.

SD-44: Airborne version with auxiliary propulsion unit which permits self-propulsion for short distances at speeds of up to 25 km/h on the road, 5.5 km/h off road.

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
85-mm HVAP-T, BR-365P/365PK
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,150
Night: INA
Armor Penetration (mm): 180 (RHA) at 1,000 meters
113 (RHA, 30°) at 500 meters

85-mm HEAT-FS, BK-2M
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,500
Night: INA
Armor Penetration (mm): 300

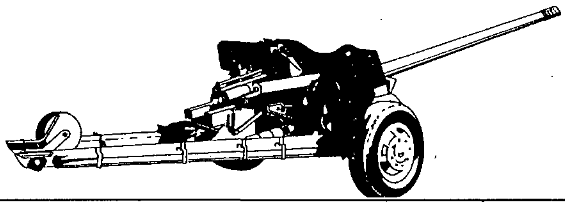
85-mm AP HE
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 950
Night: INA
Armor Penetration (mm): 91 (30° angle) at 500 meters

85-mm Frag-HE, O-365K
Maximum Aimed Range (m): 1,500
Max Effective Range (m):
Day: 1,500
Night: INA
Armor Penetration (mm): INA

Other Ammunition Types: HE, BR-365 and -365K AP-T and APC-T (obsolete)

NOTES
The gun is variously referred to as artillery, as a field gun or as an antitank gun. It can be used for all roles or specifically for artillery or antitank. Typical combat load is based on the prime mover; and a wide variety of systems can be used as prime movers.
PG-1M indirect fire sight characteristics are: 4x, 10° field of view. The PG-1 and -M can be used to a limited extent as direct fire sights.

Russian 100-mm Towed Antitank Gun MT-12



Weapons & Ammunition Types	Typical Combat Load
100-mm smoothbore gun	**20**
APFSDS-T	8
HEAT	4
Frag-HE	4
AT-10 ATGM	4

SYSTEM
Alternative Designations: T-12A, 2A29
Date of Introduction: 1972
Proliferation: At least 12 countries
Description:
Crew: 6
Combat Weight (mt): 3.1
Length Overall, Travel (m): 9.65
Height Overall, Travel (m): 1.6
Width Overall, Travel (m): 2.3

Automotive Performance:
Mount: Two-wheeled carriage with twin trails and coaster wheel
Prime mover: MT-LB-T, URAL-375D and other trucks
Towed Speed (km/h):
Max Road: 60
Max Off-Road: INA
Average Cross-Country: 25
Fording Depth (m): INA
Emplace Time (min): 2-3
Displace Time (min): 2-3
Radio: N/A
Protection: Gun shield

ARMAMENT
Main Armaments:
Caliber, Type, Name: 100-mm smoothbore gun 2A29
Rate of Fire (rd/min): 6-8/up to 15 indirect fire
Loader Type: Manual
Ready/Stowed Rounds: 0/20
Elevation (°): -7/+20
Fire on Move: No

ATGM Launcher:
Launch Method: Gun-launched, 2A29 smoothbore gun
Guidance: Laser-beam rider
Command Link: Encoded laser-beam
Launcher Dismountable: No

FIRE CONTROL
FCS Name: N/A
Main Gun Stabilization: N/A
Rangefinder: N/A
Sights w/Magnification:
Gunner:
Day: OP40M-40U direct fire, 5.5x / PG-1M indirect fire
Field of View (°): 11
Acquisition Range (m): 3,000/8,200 indirect fire
Night: APN6-40 II sight, 6.8x
Field of View (°): 7
Acquisition Range (m): 2,000

VARIANTS
T-12: Original version of Russian gun. MT-12 has changes in carriage and obturator, which do not affect lethality performance.

MT-12R: Russian upgrade with radar-directed fire control system, for use at night and adverse weather.

Topaz: Former-Yugoslav variant of T-12, with the 2A19M gun mounted on a D-30 carriage. Some have AT FCS-1 (see NOTE).

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
100-mm APFSDS-T, BM-412M, Romanian
Maximum Aimed Range (m): 2,500, 3,000 platoon volley
Max Effective Range (m):
Day: INA
Night: INA
Armor Penetration (mm): 418 at 2,000 m/380 at 3,000 m

100-mm APFSDS-T, M1000, Belgian
Maximum Aimed Range (m): 3,000/platoon volley INA
Max Effective Range (m):
Day: INA
Night: INA
Armor Penetration(mm): Triple heavy target at 4,000 meters

100-mm HEAT, BK-17
Maximum Aimed Range (m): 2,500, 3,000 platoon volley
Max Effective Range (m):
Day: INA
Night: INA
Armor Penetration (mm): 380

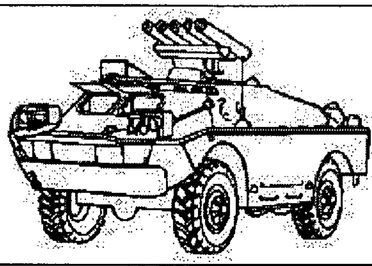
Other Ammunition Types: Russian BM-2/-20/-25 APFSDS-T; OF-15 Frag-HE; BK-5M HEAT-FS

Antitank Guided Missiles:
Name: AT-10/Kastet
Warhead Type: Shaped charge (HEAT)
Armor Penetration (mm): 650 (RHA)
Range (m): 5,000

Name: AT-10b/Kan
Warhead Type: Tandem Shaped charge (HEAT)
Armor Penetration (mm): 700 (RHA) behind ERA
Range (m): 5,000

NOTES
Russian 2nd generation II sights are available. The daysight can be used at night if the target is illuminated. Thermal sights are available. The MT-12R radar FCS can be used for surveillance, acquisition, and tracking. The Serb Iskra AT FCS-1 computerized laser rangefinder FCS is on is offered for sale. Range is 500-3,000 meters. The ATGM sight and laser guidance device has a 5,000-meter range and is a day sight only.
Ranges (m) for Frag-HE: 8,200 indirect fire/3,000 direct-fire. Rate of fire for indirect fire (Frag-HE) is up to 15 rd/min.

Russian ATGM Launcher Vehicle 9P148



Weapons & Ammunition Types	Typical Combat Load
Launcher	**15-20**
AT-5/AT-5B ATGM	15
Mixed (see NOTES)	
AT-4/AT-4B ATGM	10
AT-5/AT-5B ATGM	10

SYSTEM
Alternative Designations: BRDM-2/AT-5
Date of Introduction: 1977
Proliferation: At least 6 countries
Description:
Crew: 2
Platform: BRDM-2M/GAZ-41-08
Combat Weight (mt): 7.0
Chassis Length Overall (m): 5.73
Height (m):
Overall: 2.31
In Firing Position: INA
Width Overall (m): 2.26
Drive Formula: 4 x 4 (+ 4 auxiliary wheels)

Automotive Performance:
Engine Type: 140-hp Gasoline
Cruising Range (km): 750
Speed (km/h):
Max Road: 100
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 10
Fording Depth (m): Amphibious
Self-Entrenching Blade: N/A

Radio: R-123

Protection:
Armor, Turret Front (mm): 10
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
NBC Protection System: Collective
Smoke Equipment: N/A

ARMAMENT
Antitank Guided Missile Launcher
Name: 9P135M3 (recent upgrade)
Launch Method: tube-launched
Number of missiles on launcher: 5
Elevation (°): INA
Rate of Launch: (missiles/min): 2-3, depending on range
Reaction Time (sec): INA
Emplacement Time (min): INA
Displacement Time (min): INA
Can Launch Missiles Simultaneously : NA
Ready/Stowed Missiles: 15 (launcher + autoloader)/ 0-5 by mix
Loader Type: Automated
Launcher dismountable: No
Auxiliary Launcher: Yes
Fire on the Move: No

FIRE CONTROL
FCS Name: N/A
Guidance: SACLOS
Command Link: Wire
Beacon Type: Incandescent bulb
Tracker Type: IR, 9S451M1
Susceptible To Countermeasures: EO jammers, smoke, counterfire
Counter-countermeasures: Electro-optical jamming alarm (See note)
Rangefinder: N/A
Infrared Searchlight: N/A
Sights w/Magnification:
Gunner:
Day: 9Sh119M1
Field of View (°): INA
Acquisition Range (m): INA
Night: 1PN65
Field of View (°): INA
Acquisition Range (m): 2,500

VARIANTS
9P137: Original launcher vehicle with 5 AT-5 (only) launch rails

AMMUNITION
Antitank Guided Missiles:
Name: AT-5/SPANDREL
Alternative Designations: Konkurs
Missile Weight (kg): 25.2 (in tube)
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 650
Minimum/Maximum Range (m): 75/4,000
Probability of Hit (%): 90
Average Velocity (m/s): 200
Time of Flight to Max Range (sec): 20

Name: AT-5B
Alternative Designations: Konkurs-M
Missile Weight (kg): 26.5 (in tube)
Warhead Type: Tandem Shaped Charge (HEAT)
Armor Penetration (mm): 925
Minimum/Maximum Range (m): 75/4,000
Probability of Hit (%): 90
Average Velocity (m/s): 208
Time of Flight to Max Range (sec): 19

Name: AT-4/SPIGOT
Alternative Designations: Fagot
Missile Weight (kg): 13.0 (in tube)
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 480
Minimum/Maximum Range (m): 70/2,000
Probability of Hit (%): 90
Average Velocity (m/s): 186
Time of Flight to Max Range (sec): 11

Russian ATGM Launcher Vehicle 9P148 continued

Name: AT-4B Alternative Designations: Factoria, Konkurs M Missile Weight (kg): 13.4 (in tube) Warhead Type: Shaped Charge (HEAT) Armor Penetration (mm): 550 Minimum/Maximum Range (m): 70/2,500 Probability of Hit (%): 90 Average Velocity (m/s): 180 Time of Flight to Max Range (sec): 13.2-14.0	**Other Missile Types:** N/A

NOTES
A variety of ATGM mixes have been seen with 9P148, between AT-4 and AT-5-type ATGMS. The primary benefit of adaptability is increased launcher load and adaptability to user countries' inventories of ATGMs. Most common ATGM is AT-5. As AT-5B is produced, it is likely to replace AT-5 in better-budgeted country inventories.

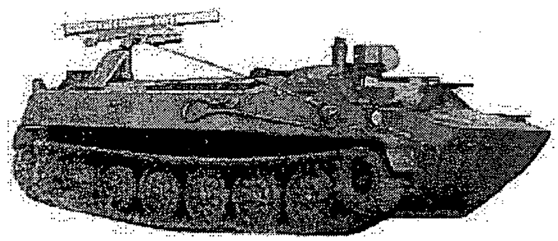
Reload time for the launcher is 25 seconds.

Russian firms have developed countermeasures, such as encoded-pulse beacons for ATGMs and counter-dazzler adjustments to the 9S451M1 guidance box. Filters can be mounted in front of reticles.

The 1PN66 thermal sight is available for the ATGM launcher. Acquisition range is approximately 2,500 meters.

Russian KBP offers a drop-in one-man turret, called Kliver, with a stabilized 2A72 30-mm gun, a 4 Kornet ATGM launcher, thermal sights, and improved fire control system.

Russian ATGM Launcher Vehicle 9P149



Weapons & Ammunition Types	Typical Combat Load
Launcher	12
AT-6 HEAT ATGM	
AT-9 HEAT ATGM	
AT-6 HE ATGM	
AT-9 HE ATGM	

SYSTEM
Alternative Designations: Shturm-S
Date of Introduction: 1990
Proliferation: At least 9 countries
Description:
Crew: 2
Platform: MT-LB
Combat Weight (mt): 12.3
Chassis Length Overall (m): 6.35
Height (m):
Overall: 1.8
In Firing Position: INA
Width Overall (m): 2.85

Automotive Performance:
Engine Type: 290-hp Diesel
Cruising Range (km): 500 km
Speed (km/h):
Max Road: 65
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 3-4
Fording Depths (m): Amphibious
Self-Entrenching Blade: Yes

Radio: R-123M or R-173

Protection:
Armor, Turret Front (mm): 7-14
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): N/A
Active Protective System: N/A
NBC Protection System: Collective
Smoke Equipment: N/A

ARMAMENT
Antitank Guided Missile Launcher
Name: INA
Launch Method: tube-launched
Number of missiles on launcher: 1
Elevation (°): -5/+15
Rate of Launch: (missiles/min): 2-3, depending on range
Reaction Time (sec): INA
Emplacement Time (min): INA
Displacement Time (min): INA
Can Launch Missiles Simultaneously: N/A
Ready/Stowed Missiles: 12/ 0
Loader Type: Automated
Launcher dismountable: No
Auxiliary Launcher: No
Fire on the Move: No

FIRE CONTROL
FCS Name: INA
Guidance: SACLOS
Command Link: Radio frequency
Beacon Type: INA
Tracker Type: IR
Susceptible To Countermeasures: Smoke, counterfire
Counter-countermeasures: 5 encoded frequencies
Rangefinder: INA
Infrared Searchlight: INA
Sights w/Magnification:
Gunner:
Day: INA
Field of View (°): INA
Acquisition Range (m): 5,000
Night: Yes
Field of View (°): INA
Acquisition Range (m): INA

VARIANTS N/A

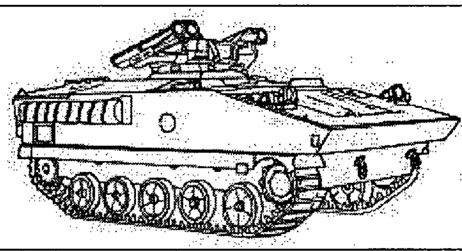
AMMUNITION
Antitank Guided Missiles
Name: AT-6a/SPIRAL
Alternative Designations: Kokon
Missile Weight (kg): 46.5 (in tube)
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 750, 600 behind ERA
Minimum/ Maximum Range (m): 400/5,000
Probability of Hit (%): 90
Average Velocity (m/s): 345
Time of Flight to Max Range (sec): 14.5

Name: AT-9
Alternative Designations: Ataka
Missile Weight (kg): 48.3 (in tube)
Warhead Type: Tandem Shaped Charge (HEAT)
Armor Penetration (mm): 950, 800 behind ERA
Minimum/Maximum Range (m): 400/6,000, 5,000 ground use
Probability of Hit (%): 90
Average Velocity (m/s): 400
Time of Flight to Max Range (sec): 15.0 (12.5 in ground use)

Other Missile Types: AT-6 HE thermobaric, AT-9 HE thermobaric

NOTES
Other missiles (AT-6b and AT-6c) can be launched from helicopters; but their length exceeds the 1832-mm limit for the Shturm-S autoloader.
A modular AT-6 ATGM launcher system with launcher and autoloader is available for installation on vehicles, fixed sites and boats.

French ATGM Launcher Vehicle AMX-10 HOT



Weapons & Ammunition Types	Typical Combat load
Total	**18**
HOT/ HOT 2, 2T/ HOT 3	

SYSTEM
Alternative Designations: INA
Date of Introduction: INA
Proliferation: At least 1 country
Description:
Crew: 4-5
Platform: AMX-10P
Combat Weight (mt): 14.1
Chassis Length Overall (m): 5.78
Height (m):
Overall: 2.57
In Firing Position: INA
Width Overall (m): 2.78

Automotive Performance:
Engine Type: 300-hp Diesel
Cruising Range (km): 600 km
Speed (km/h):
Max Road: 65
Max Off-Road: INA
Average Cross-Country: 30-40
Max Swim: 7 (with optional water jets)
Fording Depths (m): Amphibious
Self-Entrenching Blade: N/A

Radio: VHF and intercom

Protection:
Armor, Turret Front (mm): 12.7-mm frontal (distance NFI)
Applique Armor (mm): N/A
Explosive Reactive Armor (mm): Available (see NOTES)
Active Protective System: N/A
NBC Protection System: Collective
Smoke Equipment: 3 smoke grenade launchers

ARMAMENT
Antitank Guided Missile Launcher
Name: Lancelot 3
Launch Method: tube-launched
Number of missiles on launcher: 4
Elevation (°): -12/+18
Rate of Launch: (missiles/min): INA
Reaction Time (sec): INA
Emplacement Time (min): INA
Displacement Time (min): INA
Can Launch Missiles Simultaneously : INA
Ready/Stowed Missiles: 4/14
Loader Type: Manual
Launcher dismountable: No
Auxiliary Launcher: No
Fire on the Move: No

FIRE CONTROL
FCS Name: INA
Guidance: SACLOS
Command Link: Wire
Beacon Type: INA
Tracker Type: INA
Susceptible To Countermeasures: Smoke, counterfire
Counter-countermeasures: Infrared CM hardening on later ATGMs
Rangefinder: M427 Laser rangefinder
Infrared Searchlight: INA
Sights w/Magnification:
Gunner:
Day: M509, 3x/12x
Field of View (°): INA
Acquisition Range (m): INA
Night: Castor Thermal Image System available
Field of View (°): INA
Acquisition Range (m): INA

VARIANTS N/A

Antitank Guided Missiles
Name: HOT
Alternative Designations: Euromissile
Missile Weight (kg): 32 (in tube)
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 800
Minimum/ Maximum Range (m): 75/4,000
Probability of Hit (%): INA
Average Velocity (m/s): 233
Time of Flight to Max Range (sec): 17.3

Name: HOT 2
Alternative Designations: INA
Missile Weight (kg): 32 (in tube)
Warhead Type: Tandem Shaped Charge (HEAT)
Armor Penetration (mm): 900
Minimum/Maximum Range (m): 75/4,000
Probability of Hit (%): INA
Average Velocity (m/s): 233
Time of Flight to Max Range (sec): 17.3

Name: HOT 2T
Alternative Designations: INA
Missile Weight (kg): 32 (in tube)
Warhead Type: Tandem shaped Charge (HEAT)
Armor Penetration (mm): 1250
Minimum/Maximum Range (m): 75/4,000
Probability of Hit (%): INA
Average Velocity (m/s): INA
Time of Flight to Max Range (sec): INA

Other Missile Types: HOT 3--similar to HOT 2T, but with improved countermeasures

French ATGM Launcher Vehicle AMX-10 HOT continued

NOTES

The HOT Antitank guided missile is produced by a European consortium which includes France and Germany. It can be launched from a ground launcher, the same launcher mounted on a variety of vehicles, from infantry fighting vehicles and ATGM launcher vehicles, and from helicopters. The AMX-10 HOT constitutes a high-end application on that spectrum, and has not been widely proliferated.

The cruciform-based single-tube ground launcher system exceeds the weight limit for the portable class of ATGM launchers. An updated launcher for HOT-2T offers a Thermal Modular System night sight and a dual band tracker. Alternate mounts for the launcher include the ATLAS/Commando lightweight launcher (140 kg) mounted on the Spanish Santana (4 x 4 Land Rover light truck).

The Lancelot turret used on AMX-10 HOT can be mounted on other armored fighting vehicles.

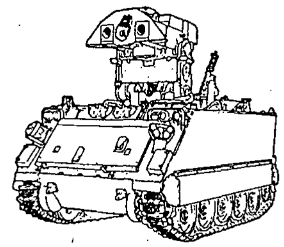
The French-produced VAB HOT uses a Mephisto retractable twin-tube launcher, and has an onboard load of 10 HOT ATGMs.

The UTM800 turret holds four HOT missiles, with a stabilized sight and Castor thermal night sight. The UTM800 is used on two applications. The French VCR/TH employs the turret on a Panhard VCR/TT 6 x 6 APC chassis. The other is the UTM turret on a VAB APC chassis.

The German Jaguar 1 Jagdpanzer is a modified Leopard 1 tank chassis with a single-tube HOT launcher.

French SNPE explosive reactive armor can be employed on AMX-10 type vehicles.

US ATGM Launcher Vehicle M901



Weapons & Ammunition Types	Typical Combat Load
ATGM Launcher TOW, ITOW, TOW 2, TOW 2A, TOW 2B	12
7.62-mm Cupola MG	2,000

SYSTEM
Alternative Designations: ITV (Improved TOW Vehicle), ITOW
Date of Introduction: 1978
Proliferation: At least 8 countries
Description:
Crew: 4-5
Platform: M113A1
Combat Weight (mt): 11.79
Chassis Length Overall (m): 4.90
Height (m):
Overall: 2.91
In Firing Position: 3.35
Width Overall (m): 2.70

Automotive Performance:
Engine Type: 212-hp Diesel
Cruising Range (km): 483
Speed (km/h):
Max Road: 64
Max Off-Road: INA
Average Cross-Country: INA
Max Swim: 5.8
Fording Depths (m): Amphibious
Self-Entrenching Blade: N/A

Radio: Various, including intercom

Protection:
Armor, Turret Front (mm): INA
Applique Armor (mm): Available. Anti-mine armor on bottom
Explosive Reactive Armor (mm): Available
Active Protective System: No
NBC Protection System: No
Smoke Equipment: 4 smoke grenade launchers on each front corner

ARMAMENT
Antitank Guided Missile Launcher
Name: M27 cupola with launcher head ("Hammerhead")
Launch Method: Tube-launched
Number of missiles on launcher: 2
Elevation (°): -30/+34
Rate of Launch: (missiles/min): 2
Reaction Time (sec): 4.25
Emplacement Time (min): 0.33
Displacement Time (min): INA
Can Launch Missiles Simultaneously : No
Ready/Stowed Missiles: 2/10
Loader Type: Manual
Launcher dismountable: No
Auxiliary Launcher: No
Fire on the Move: No

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm (7.62x51) MG
Mount Type: Cupola
Direct Fire Range (m): INA
Max Effective Range (m):
Day: INA
Night: INA
Fire on Move: Yes
Rate of Fire: INA

Firing Ports: INA

FIRE CONTROL
FCS Name: INA
Guidance: SACLOS
Command Link: Wire
Beacon Type: Xenon (Infrared), thermal on TOW-2 and after
Tracker Type: INA
Susceptible To Countermeasures: Smoke, counterfire
Counter-countermeasures:
Rangefinder: INA
Infrared Searchlight: INA
Sights w/Magnification:
Gunner:
Day: Day sight/tracker, 13x
Field of View (°): 5.5 x
Acquisition Range (m): INA
Night: AN/TAS-4 thermal sight
Field of View (°): INA
Acquisition Range (m): INA

VARIANTS
ITOW: Launcher variants have been upgraded with new turrets and launcher heads to fit the later TOW variants, such as ITOW, TOW 2, 2A and 2B. **M901A2:** Launcher vehicle fitted for TOW 2.

A variety of M113-based vehicles have incorporated TOW "hammerhead" launcher for use as ATGM launcher vehicles. These include the Italian VCC-1-based launcher vehicle, and the Dutch Armored Infantry Fighting Vehicle (AIFV) -based launcher vehicle.

AMMUNITION
Antitank Guided Missiles
Name: TOW
Alternative Designations: BGM-71
Missile Weight (kg): 25.5 (in tube)
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 600
Minimum/ Maximum Range (m): 65/3,750
Probability of Hit (%): INA
Average Velocity (m/s): 179
Time of Flight to Max Range (sec): 21

US ATGM Launcher Vehicle M901 continued

Name: ITOW Alternative Designations: BGM-71C Missile Weight (kg): 25.7 (in tube) Warhead Type: Tandem Shaped Charge (HEAT, short probe) Armor Penetration (mm): 800 Minimum/ Maximum Range (m): 65/3,750 Probability of Hit (%): INA Average Velocity (m/s): 179 Time of Flight to Max Range (sec): 21 Name: TOW 2 Alternative Designations: BGM-71D Missile Weight (kg): 28.1 (in tube) / 21.5 (missile only) Warhead Type: Tandem Shaped Charge (Larger HEAT, long probe) Armor Penetration (mm): INA Minimum/ Maximum Range (m): 65/3,750 Probability of Hit (%): 90 Average Velocity (m/s): 179 Time of Flight to Max Range (sec): 21	Name: TOW 2A Alternative Designations: BGM-71E Missile Weight (kg): 22.65 (missile only) Warhead Type: Tandem Shaped Charge (Larger HEAT, long probe) Armor Penetration (mm): INA Minimum/ Maximum Range (m): 65/3,750 Probability of Hit (%): INA Average Velocity (m/s): 188 Time of Flight to Max Range (sec): 20 Name: TOW 2B Alternative Designations: BGM-71F Missile Weight (kg): 22.60 (missile only) Warhead Type: Dual explosive-formed penetrators (EFP), top-attack Armor Penetration (mm): INA Minimum/ Maximum Range (m): 200/3,750 Probability of Hit (%): INA Average Velocity (m/s): 179 Time of Flight to Max Range (sec): 21 **Other Missile Types:** See NOTES, below

NOTES

The loader has side and overhead protection during loading, which requires 40 seconds.

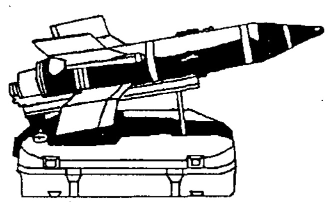
The Improved Target Acquisition System (ITAS) was developed for TOW 2 and later. It includes a laser rangefinder, increased acquisition range, improved night capabilities (second-generation thermal channel), an automatic boresight and greater hit probability.

The UK-developed Further-Improved TOW (FITOW) program is expected to be similar to TOW 2B, but with two smaller warheads.

The Israeli MAPATS is a TOW missile variant with laser-beam rider guidance and a laser guidance system.

The Israeli TAAS tandem warhead is the same diameter as the warhead on the original TOW missile, and appears to be a candidate for retrofit. The warhead is claimed to be able to penetrate 1,020 mm of armor.

Russian ATGM Launcher AT-3



Weapons & Ammunition Types	Typical Combat Load
ATGM Launcher	**4/3 Polk Set**
AT-3 HEAT ATGM	1 on launcher
AT-3 HE ATGM	

SYSTEM
Alternative Designations: Malyutka Complex
Date of Introduction: 1963
Proliferation: At least 45 countries
Description:
Crew: 3
Primary Mount: Ground mount on "suitcase" launcher
Alternate Mounts: Rail on BMP-1, BMD-1, BRDM, BRDM-2 etc.
Weight Overall, Excluding Missile (kg): 30.5 launcher + guidance
Length Overall in Firing Position (m): 0.86 with AT-3/a/b/c
1.02 with Malyutka-2
Height Overall In Firing Position (m): INA
Width Overall In Firing Position (m): INA

ARMAMENT
Launcher
Name: 9P111 Case launcher
Launch Method: Rail on case
Elevation (°): Fixed for launcher (see NOTES)
Rate of Launch: (missiles/min): 2
Reaction Time (sec): INA
Emplacement Time (min): 1.7 POLK set
Displacement Time (min): INA
Ready/Stowed Missiles: 4/0, 3/0 POLK set

FIRE CONTROL
FCS Name: 9S415/9S415M/9S415M1 guidance panel
Guidance: MCLOS (9S415/-M panel), SACLOS
Command Link: Wire
Beacon Type: Incandescent infrared bulb (SACLOS)
Tracker Type: N/A for MCLOS, flare tracker for SACLOS
Susceptible To Countermeasures: EO jammers, smoke, counterfire
Counter-countermeasures: Offset guidance panel, laser filters
Rangefinder: INA
Frequency: INA
Counter-countermeasures: INA
Sights w/Magnification:
Gunner:
Day: 9Sh16, 8x
Field of View (°): 22.5 (see NOTES)
Acquisition Range (m): 4000
Night: Available
Field of View (°): N/A
Acquisition Range (m): N/A

VARIANTS
Copies include North Korean Susong-Po, Taiwanese Kun Wu, and the Chinese copy, Red Arrow-73/HJ-73, with indigenous guidance.

POLK: Slovenian Portable Anti-armor Launching Set includes a new launcher, guidance panel with binocular sight, and 3 ATGMs similar to AT-3C Improved (nose probes and lower smoke signature).
With a nose probe and improved propellant, the MCLOS-guided ATGM can reach maximum range in 25 sec and penetrate 580 mm.
A Russian AT-3c/Improved (SACLOS) has similar capabilities.

AMMUNITION
Antitank Guided Missiles
Name: AT-3, -3a, -3b/SAGGER
Alternative Designations: Malyutka, Malyutka-M
Missile Weight (kg): 10.9
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 400
Minimum/Maximum Range (m): 500/3,000
Probability of Hit (%): 70 against moving tanks
Average Velocity (m/s): 115
Time of Flight to Max Range (sec): 26

Name: AT-3c/SAGGER
Alternative Designations: Malyutka-P
Missile Weight (kg): 11.4
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 520
Minimum/Maximum Range (m): 500/3,000
Probability of Hit (%): 90 (SACLOS)
Average Velocity (m/s): 115
Time of Flight to Max Range (sec): 26

Name: Malyutka-2
Alternative Designations: Malyutka (Modernized)
Missile Weight (kg): 12.5
Warhead Type: Tandem Shaped Charge (HEAT)
Armor Penetration (mm): 800
Minimum/Maximum Range (m): 500/3,00
Probability of Hit (%): 90 (SACLOS)
Average Velocity (m/s): 130
Time of Flight to Max Range (sec): 23

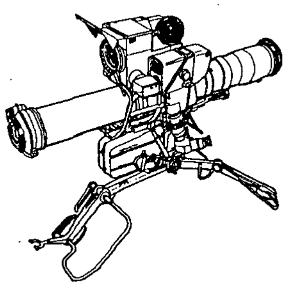
Other Missiles: Malyutka (Modernized) HE, AT-3c Imp, POLK

NOTES
AT-3 is classed by weight as portable (21-40 kg), rather than manportable (<21 kg). The launcher is also a missile carry case. The guidance panel can be located up to 15 meters from the launcher, and can control up to four launchers. If target is <1,000 meters from launcher, the operator can joystick the missile to target without using optics. Guidance elevation (°) is -5/ +10. Because the module is small and can be shifted, elevation and field of view are operationally unlimited. Improved versions can be used on older launchers, but in the MCLOS mode.

The Slovenian Iskra TS-M thermal sight is available, with detection at 3,000 meters and recognition at 1,800 meters.

Any AT-3 can be modernized to Malyutka-2 with replacement of warhead and or replacement of specific warhead and motor components.

Russian ATGM Launcher AT-4/AT-5



9P135M3 w/AT-5B and thermal sight

Weapons & Ammunition Types	Typical Combat Load
ATGM Launcher Total	**4 or 8**
AT-4/AT-4B ATGM	(see NOTES)
AT-5/AT-5B ATGM	

SYSTEM
Alternative Designations: 9P135M Firing Post, Fagot/Fagot-M
Date of Introduction: 1973
Proliferation: At least 25 countries
Description:
Crew: 3
Primary Mount: Ground mount on folding tripod
Alternate Mounts: Pintel (post) on BMP-1P, BTR-D, UAZ-469, etc.
Weight Overall, Excluding Missile (kg): 22.5
Length Overall in Firing Position (m): 1.1/1.3 AT-4/5 tube
Height Overall In Firing Position (m): INA
Width Overall In Firing Position (m): INA

ARMAMENT
Launcher
Name: 9P135 (AT-4 only), 9P135M (AT-4/AT-5), -M1, -M2, -M3
Launch Method: Tube-launched
Elevation (°) (-/+): INA
Rate of Launch: (missiles/min): 2-3, depending on range
Reaction Time (sec): INA
Emplacement Time (min): INA
Displacement Time (min): INA
Ready/Stowed Missiles: 4/0 full dismount, 4/4 on or near vehicle

FIRE CONTROL
FCS Name: 9S451M1 Guidance control box
Guidance: SACLOS
Command Link: Wire
Beacon Type: Incandescent infrared bulb
Tracker Type: IR, 9S451M1
Susceptible To Countermeasures: EO jammers, smoke, counterfire
Counter-countermeasures: EO jamming alarm (see NOTES)
Rangefinder: INA
Sights w/Magnification:
Gunner:
Day: 9Sh119M1, 4x
Field of View (°): 4.5
Acquisition Range (m): INA
Night: Available (See NOTES)

VARIANTS
P135M3: Konkurs-M Complex. Launcher with 1PN65 thermal sight and AT-5B/Konkurs-M missiles. Night range is 2,500m.

AMMUNITION
Antitank Guided Missiles
Name: AT-5B/SPANDREL-B
Alternative Designations: Konkurs-M
Missile Weight (kg): 26.5 (in tube)
Warhead Type: Tandem Shaped Charge (HEAT)
Armor Penetration (mm): 925
Minimum/Maximum Range (m): 75/4,000
Probability of Hit (%): 90
Average Velocity (m/s): 208
Time of Flight to Max Range (sec): 19

Name: AT-5/SPANDREL
Alternative Designations: Konkurs
Missile Weight (kg): 25.2 (in tube)
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 650
Minimum/Maximum Range (m): 75/4,000
Probability of Hit (%): 90
Average Velocity (m/s): 200
Time of Flight to Max Range (sec): 20

Name: AT-4/SPIGOT
Alternative Designations: Fagot
Missile Weight (kg): 13.0 (in tube)
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 480
Minimum/Maximum Range (m): 70/2,000
Probability of Hit (%): 90
Average Velocity (m/s): 186
Time of Flight to Max Range (sec): 11

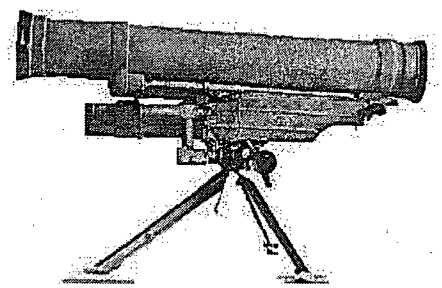
Other Missiles: AT-4B/Factoria (see NOTES)

NOTES
Because of its weight, the Russians categorize the AT-4/4B system as portable (21-40 kg) rather than manportable. For dismounted carry load is divided among three packs. Due to the greater weight, AT-5/-5B fits into the "heavy" class (40+ kg), and should only be carried short distances from vehicles (<500 meters). For crews using both ATGM classes and operating near vehicles, combat load is 8 (4 stowed in the vehicle).

The AT-4B/Factoria is an upgrade ATGM with a 2,500 meter range, 550-mm penetration, and a velocity of 180 m/s (13.2 - 14.0 sec TOF).
Russian firms have developed counter-countermeasures, such as encoded-pulse beacons for ATGMs and counter-dazzler adjustments to the 9S451M1 guidance box. Filters can be mounted in front of reticles.

TPVP/1PN65 thermal sight is available, with the range approximately 2,500 meters (see VARIANTS, above). Weight is 13 kg. Slovenian TS-F sight and Russian 1PN86-1/1PN86/Mulat have a 3,600 meter detection range.

Russian ATGM Launcher AT-7/AT-13



Weapons & Ammunition Types	Typical Combat Load
ATGM Launcher	4
AT-7 HEAT ATGM	
AT-13 HEAT ATGM	
AT-13 HE ATGM	

SYSTEM
Alternative Designations: 9P151Firing Post
Date of Introduction: 1978
Proliferation: At least 5 countries

Description:
Crew: 2
Primary mount: Ground mount on tripod
Alternate mounts: Shoulder for launch, UAZ-469 pintel mount
Weight Overall, Excluding Missile (kg): 10.2
Length Overall in Firing Position (m): 0.78 with AT-7/Metis
0.98 with AT-13/Metis-M
Height Overall In Firing Position (m): 0.72 with AT-7/Metis
Width Overall In Firing Position (m): INA

ARMAMENT
Launcher
Name: 9P151 Firing Post
Launch Method: Tube
Elevation (°): -5/+10
Rate of Launch (missiles/min): 3-5, depending on range
Reaction Time (sec): INA
Emplacement Time (min): 0.20
Displacement Time (min): 0.33
Ready/Stowed Missiles: 4/0 (1 on launcher)

FIRE CONTROL
FCS Name: 9S816 Guidance system
Guidance: SACLOS
Command Link: Wire
Beacon Type: INA
Tracker Type: IR
Susceptible To Countermeasures: EO jammers, smoke, counterfire
Counter-countermeasures: INA

Rangefinder:
Frequency: INA
Counter-countermeasures: INA

Sights w/Magnification:
Gunner:
Day: INA
Field of View (°): INA
Acquisition Range (m): INA
Night: Available

VARIANTS
Metis-M System: 9P151firing post adapted for and including the Metis-M missile, IOC 1992.

AMMUNITION
Antitank Guided Missiles
Name: AT-7/Saxhorn
Alternative Designations: Metis
Missile Weight (kg): 6.3 (in tube)
Warhead Type: Shaped Charge (HEAT)
Armor Penetration (mm): 460
Minimum/Maximum Range (m): 40/1,000
Probability of Hit (%): 90
Average Velocity (m/s): 180
Time of Flight to Max Range (sec): 6.2

Name: AT-13
Alternative Designations: Metis-M (often mislabeled Metis-2)
Missile Weight (kg): 13.8 (in tube)
Warhead Type: Tandem Shaped Charge (HEAT)
Armor Penetration (mm): 1,000/900 behind ERA
Minimum/Maximum Range (m): 80/1500
Probability of Hit (%): 90
Average Velocity (m/s): 287
Time of Flight to Max Range (sec): 8

Other missiles: Metis-M HE thermobaric

NOTES
The Russians characterize the AT-7 ATGM complex as light or manportable (5-20 kg), permitting long-distance carry by dismounted infantry. Although the AT-13 complex slightly exceeds 20 kg, it is close enough to fit into the category.

Guidance elevation has a 15° span. Because the module is small and can be quickly corrected by shifting, elevation and field of view are operationally unlimited, and permit use against hovering or stationary helicopters.

The Russian 1PN86V/Mulat-115 thermal sight is available for use on the launcher, with detection at 3,200 meters and recognition beyond the missile's 1,500 meter range. Field of view is 4.6°.

French ATGM Launcher Eryx

	Weapons & Ammunition Types	Typical Combat Load
	**ATGM Launcher** Eryx ATGM	1

SYSTEM
Alternative Designations: Anti-Char Courtee Portee (ACCP)
Date of Introduction: 1991
Proliferation: At least 5 countries
Description:
Crew: 1
Primary mount: Ground mount on tripod or shoulder launch
Alternate mounts: Shoulder launch--standing, kneeling or prone
Weight Overall, Excluding Missile (kg): 3, 4 with tripod
Length Overall in Firing Position (m): 0.905
Height Overall In Firing Position (m): INA
Width Overall In Firing Position (m): INA tripod, 0.16 on shoulder

ARMAMENT
Launcher
Name: Eryx
Launch Method: Tube (disposable canister/ launch tube)
Elevation (°): INA, tripod; unlimited on shoulder launch
Rate of Launch: (missiles/min): INA
Reaction Time (sec): 20-30 (includes emplace time)
Emplacement Time (min): See Reaction Time (above)
Displacement Time (min): < 0.03
Ready/Stowed Missiles: 1/ 0

FIRE CONTROL
FCS Name: INA
Guidance: SACLOS
Command Link: Wire
Beacon Type: Infrared laser diode
Tracker Type: Charged couple device (CCD)
Susceptible To Countermeasures: EO jammers, smoke, counterfire
Counter-countermeasures: Flight time less than 4 seconds

Rangefinder: INA
Sights w/Magnification:
Gunner:
Day: INA, 3x
Field of View (°): 3.4
Acquisition Range (m): INA
Night: Sopelem OB50 II sight
Field of View (°): INA
Acquisition Range (m): INA

VARIANTS
N/A

AMMUNITION
Antitank Guided Missile
Name: Eryx
Alternative Designations: ACCP
Missile Weight (kg): 11 (in tube)
Warhead Type: Tandem Shaped Charge (HEAT)
Armor Penetration (mm): 900
Minimum/Maximum Range (m): 50/600
Probability of Hit (%): 90
Average Velocity (m/s): 162
Time of Flight to Max Range (sec): 3.7

Other missiles: N/A

NOTES
The disposable canister/launch tube is attached to the reusable firing post (which includes sight systems).

Eryx employs a recoil reduction system with reduced back-blast, which permits launch from inside of buildings. Signature reduction includes noise and smoke reduction.

A rest such as a ledge or sandbag is required for launches beyond 350 meters.

The optional French Mirabel thermal night sight is available for use on Eryx. The Mirabel offers an acquisition range of 1,000 meters, but weighs an additional 3.4 kg.

Chapter 6
Artillery

This chapter provides the basic characteristics of selected artillery weapon systems either in use or readily available to the OPFOR. Therefore, the artillery systems discussed in this chapter are those likely to be encountered by U.S. forces in varying levels of conflict. The selection of artillery systems is not intended to be all-inclusive, rather a representative sampling of weapons and equipment supporting various military capabilities.

This chapter is divided into the following categories—artillery reconnaissance, towed artillery systems, self-propelled artillery systems, and multiple rocket launchers. Later updates of this guide will include data sheets addressing the aforementioned categories as well as mortars, artillery locating radars, sound and flash systems, and surface to surface missiles (SSMs).

OPFOR artillery units begin a battle with a full complement of ammunition to include special types of ammunition. The number and type of rounds vary according to the tactical situation and mission. Therefore, we have used frag-HE, smoke, and illumination as the default rounds to represent a typical combat load. Generally, the Typical Combat Load section represents the number and type of rounds carried on the self-propelled artillery system or rocket launcher. The numbers of rounds for the towed artillery systems vary according to the cargo capacity of the prime mover.

Questions and comments on data listed in this chapter should be addressed to:

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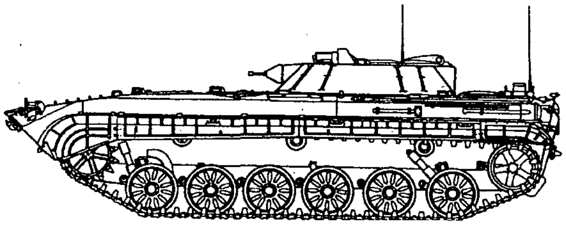
Russian Artillery Mobile Reconnaissance Vehicle PRP-3/PRP-4M



Weapons & Ammunition Types	Typical Combat Load
7.62 PKT MG	2,000

SYSTEM
Alternative Designations: None
Date of Introduction: 1975
Proliferation: At least 1 country
Description:
Crew: 5
Platform (chassis): BMP-1
Combat Weight (mt): 13.2
Chassis Length Overall (m): 6.73
Height Overall (m): 2.14
Width Overall (m): 2.94

Automotive Performance:
Engine Type: 293 hp Diesel
Cruising Range (km): 600 km
Speed (km/h):
Max Road: 60
Max Off-Road: 35
Cross-Country: INA
Max Swim: 7
Fording Depths (m): Amphibious

Radio: R-173

Protection:
Armor, Turret (mm): 23
Armor Hull (mm): 19
Self-Entrenching Blade: No
NBC Protection System: Yes
Smoke Equipment: Vehicle engine exhaust smoke system (VEESS)

ARMAMENT
Main Armament:
Caliber, Type, Name: 7.62-mm machinegun PKT
Mount Type: coax
Direct Fire Range (m): 1300
Max Effective Range (m):
Day: 1000 / 400-500 on the move
Night: 800
Fire on Move: Yes
Rate of Fire (rpm): 600 cyclic in 2-10 round bursts

VARIANTS
None

SENSORS/COMPONENTS

PRP-3 Sensors/Components:
navigation: 1G25 gyrocompass and 1G13 gyro course indicator
fire direction: 1V520 Ballistic Computer

right side sensors: 1PN61 Night Vision sensor and 1D11 Laser Range-finder

left side sensors: none

Radar: 1RL126 Small Fred Radar
operating band: K (36.2 – 37.0 GHz)
detection range: 20 km
tracking range: 7 – 12 km

PRP-4 Sensors/Components:
navigation: 1G25-1 gyrocompass and 1G13 gyro course indicator
fire direction: 1V520 Ballistic Computer

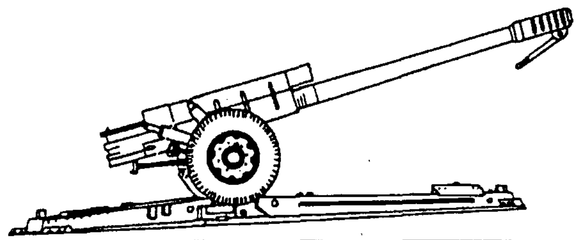
right side sensors: 1PN61 Night Vision sensor and 1D11M-1 Laser Rangefinder

left side sensors: 1PN59 Thermal Imaging Night Vision Device and 1D14 Laser Rangefinder

Radar: 1RL133M-1 Tall Mike Radar
operating band: I (9.0 GHz)
detection range (personnel): 3.0 km
detection range (vehicle): 12 km

NOTES
The PRP-4M has improved 1PN71 night vision sensors. The vehicles are also equipped with a NBC filtration and overpressure system.

Russian 122-mm Towed Howitzer D-30A



Weapons & Ammunition Types	Typical Combat Load
122-mm howitzer	
Frag-HE	
Smoke	
Illumination	

SYSTEM
Alternative Designations: 122-mm D-30A Lyagushka
Date of Introduction: 1963
Proliferation: At least 13 countries
Description:
Crew: 5 (section of 6)
Carriage: D-30
Combat Weight (mt): 3.2
Chassis Length Overall (m):
  Travel Position: 5.4
  Firing Position: INA
Height Overall (m): 1.6
Width Overall (m):
  Travel Position: 1.9
  Firing Position: INA
Towing Speed (km/h):
  Max Road: 60
  Max Off-Road: 25
  Max Cross-Country:
  Fording Depths (m): .5
Emplacement Time (min): 1.5
Displacement Time (min): 3.5

Prime Mover: MT-LB; Ural-375, or equivalent

ARMAMENT
Main Armament:
Caliber, Type, Name: 122-mm, 2A18M canon
Barrel Length (cal): 38 (approximately)
Rate of Fire (rpm):
  Burst: 8
  Normal: 6
  Sustained: 4
Loader Type: Semi-automatic
Breech Type: Vertical sliding wedge
Muzzle Brake Type: Multi-baffle
Traverse: (°):
  Left: 360
  Right: 360
  Total: 360
Elevation (°) (-/+): -7/+70°

FIRE CONTROL
Indirect Fire: PG-1M Panoramic Telescope (PANTEL)
Direct Fire: OP 4M-45
Collimator: K-1
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
Saddam: Iraqi produced version of the D-30
D30J: Yugoslavian produced version of the D-30
SP 122: Egyptian self-propelled howitzer (M109A2 chassis and turret with an Egyptian made D-30 howitzer).
Type 85: Chinese self-propelled howitzer (Chinese Type 85 APC chassis and a licensed produced version of the D-30 in a semi-open superstructure.)

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
122-mm Frag-HE, OF-81
  Indirect Fire Range (m):
    Minimum Range: 1000
    Maximum Range: 15,300
  Complete Projectile Weight (kg): 21.76 (OF-56)
  Muzzle Velocity: 680 m/s
  Fuze Type: RGM-2 PD

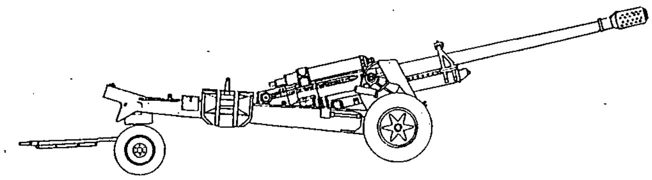
122-mm, HEAT-FS
  Direct Fire Range (m):
    Minimum Range: 0
    Maximum Range: 1000
  Armor Penetration (mm): 460 (@ 0° obliquity any range)
  Complete Projectile Weight (kg): 21.58
  Muzzle Velocity: 740 m/s
  Fuze Type: GPV-2 PIBD

122-mm Frag-HE Rocket Assisted
  Indirect Fire Range (m):
    Minimum Range: INA
    Maximum Range: 21,900
  Complete Projectile Weight (kg): 21.76 (3OF-56)
  Muzzle Velocity: INA
  Fuze Type: PD

Other Ammunition Types: Incendiary, Chemical, Flechette, Semi-active laser-guided Kitolov-2M Frag-HE

NOTES
The D-30A is a midlife product improvement of the D-30. The original D-30 was fielded in 1963 and the midlife product improvements occurred in the mid to late 1970's. The original D-30 is in use with at least 50 different countries.

Russian 130-mm Towed Gun M-46



Weapons & Ammunition Types	Typical Combat Load
130-mm howitzer	
Frag-HE	
Smoke	
Illumination	

SYSTEM
Alternative Designations: None
Date of Introduction: 1954
Proliferation: At least 25 countries
Description:
Crew: 8
Carriage: M-46
Combat Weight (mt): 8.45
Chassis Length Overall (m):
Travel Position: 11.73
Firing Position: 11.10
Height Overall (m): 2.55
Width Overall (m):
Travel Position: 2.45
Firing Position: INA
Towing Speed (km/h):
Max Road: 50
Max Off-Road: 20
Max Cross-Country: 10
Fording Depths (m): INA
Emplacement Time (min): 6
Displacement Time (min): 7

Prime Mover: AT-S 59, KrAZ-255 or equivalent

ARMAMENT
Main Armament:
Caliber, Type, Name: 130-mm, canon
Barrel Length (cal): 52 (approximately)
Rate of Fire (rpm):
Burst: 8
Normal: 6
Sustained: 5
Loader Type: Manual
Breech Type: Horizontal sliding wedge
Muzzle Brake Type: Multiperforated (pepperpot)
Traverse: (°):
Left: 25
Right: 25
Total: 50
Elevation (°) (-/+): -2.5/+45°

FIRE CONTROL
Indirect Fire: PG-1 Panoramic Telescope (PANTEL)
Direct Fire: OP 4-35
Collimator: K-1
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
None

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
130-mm Frag-HE, OF44
Indirect Fire Range (m):
Minimum Range: INA
Maximum Range: 22,500
Complete Projectile Weight (kg): 33.40 (OF33)
Muzzle Velocity: 930 m/s
Fuze Type: V-429 PD

130-mm, APC-T
Direct Fire Range (m):
Minimum Range: 0
Maximum Range: 1140
Armor Penetration (mm): INA
Complete Projectile Weight (kg): 33.49 (BR-482B)
Muzzle Velocity: INA
Fuze Type: DBR BD

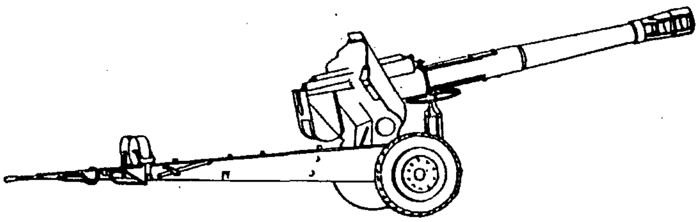
130-mm Frag-HE, OF-43
Indirect Fire Range (m):
Minimum Range: INA
Maximum Range: 27,500
Complete Projectile Weight (kg): 33.40 (OF-33)
Muzzle Velocity: 930 m/s
Fuze Type: V-429 PD

130-mm Frag-HE, ERFB-BB
Indirect Fire Range (m):
Minimum Range: INA
Maximum Range: 38,000
Complete Projectile Weight (kg): 33.40
Muzzle Velocity: 940 m/s
Fuze Type: ML-5 PD

Other Ammunition Types: Smoke, Chemical, Illumination

NOTES:
The M-46 gun crew is provided limited frontal protections by virtue of a frontal V-shaped shield (approximately 7-mm thick). Otherwise, the crew, ammunition supply, and equipment are vulnerable to casualties and damage from small arms fire, artillery fire, and bomb shrapnel. The Extended Range Full Bore-Base Bleed round was specifically designed by NORINCO Industries (China) for use with the Chinese 130-mm Type 59 Field Gun. However, this round may be fired by the M-46.

Russian 152-mm Towed Gun-Howitzer D-20



Weapons & Ammunition Types	Typical Combat Load
152-mm howitzer	
Frag-HE	
Smoke	
Illumination	

SYSTEM
Alternative Designations: None
Date of Introduction: 1955
Proliferation: At least 13 countries
Description:
Crew: 8
Carriage: 122-mm gun D-74
Combat Weight (mt): 5.7
Chassis Length Overall (m):
  Travel Position: 8.10
  Firing Position: 8.69
Height Overall (m): 2.52
Width Overall (m):
  Travel Position: 2.35
  Firing Position: INA
Towing Speed (km/h):
  Max Road: 60
  Max Off-Road: 30
  Max Cross-Country: 15
Fording Depths (m): .5
Emplacement Time (min): 2.5
Displacement Time (min): 2.5

Prime Mover: AT-S Tracked vehicle; MT-LB; Ural-375; Ural-4320

ARMAMENT
Main Armament:
Caliber, Type, Name: 152-mm, canon
Barrel Length (cal): 25
Rate of Fire (rpm):
  Burst: 5-6
  Normal: INA
  Sustained: 1 (65 rounds the first hour)
Loader Type: Manual
Breech Type: Vertical sliding wedge
Muzzle Brake Type: Double flared
Traverse: (°):
  Left: 29
  Right: 29
  Total: 58
Elevation (°) (-/+):-5/+45°

FIRE CONTROL
Indirect Fire: PG-1M Panoramic Telescope (PANTEL)
Direct Fire: OP 4M
Collimator: K-1
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
None

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
152-mm Frag-HE, OF32
  Indirect Fire Range (m):
    Minimum Range: 4600
    Maximum Range: 17,400
  Complete Projectile Weight (kg): 43.56 (OF25)
  Muzzle Velocity: 655 m/s
  Fuze Type: V-90 PD

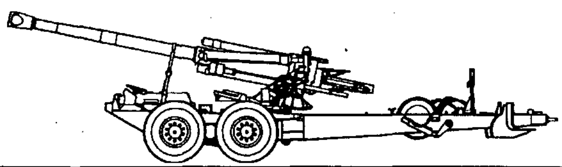
152-mm, HEAT, BP-540
  Direct Fire Range (m):
    Minimum Range: 0
    Maximum Range: 1000
  Armor Penetration (mm): INA
  Complete Projectile Weight (kg): 27.00
  Muzzle Velocity: 655 m/s
  Fuze Type: GPV-3 PD

152-mm Frag-HE, OF-96
  Indirect Fire Range (m):
    Minimum Range: INA
    Maximum Range: 24,400
  Complete Projectile Weight (kg): 43.56 (OF-64)
  Muzzle Velocity: INA
  Fuze Type: PD

Other Ammunition Types: DPICM, DPICM-BB, Incendiary, Expendable Jammer, Chemical, Flechette, Semi-active laser-guided Krasnopol-M Frag-HE

NOTES
The D-20 was the first 152-mm cannon system to incorporate a semiautomatic vertical-sliding-wedge breech block. Although the ammunition for the system was not changed, this modification allowed a slightly higher rate of fire to be achieved (6 rounds per minute rather than 4), although the sustained rate of fire was unchanged. Because the carriage is based on that of the 122-mm gun D-74, the D-20 cannot be elevated above 45°.

South African 155-mm Towed Gun-Howitzer G5

	Weapons & Ammunition Types	Typical Combat Load
	155-mm howitzer	
	Frag-HE Smoke Illumination	

SYSTEM
Alternative Designations: None
Date of Introduction: 1981
Proliferation: At least 4 countries
Description:
Crew: 8
Carriage: G5
Combat Weight (mt): 13.75
Chassis Length Overall (m):
- Travel Position: 12.1
- Firing Position: 11.0

Height Overall (m): 2.3
Width Overall (m):
- Travel Position: 3.3
- Firing Position: 8.7

Towing Speed (km/h):
- Max Road: 90
- Max Off-Road: 50
- Max Cross-Country: 15

Fording Depths (m): .6
Emplacement Time (min): 2
Displacement Time (min): 1

Auxiliary Propulsion Unit Performance:
Engine Type: 76 hp air-cooled diesel
Cruising Range (km): 100
Speed (km/h):
- Max Road: 16
- Max Off-Road: INA
- Cross-Country: 3
- Max Swim: N/A

Prime Mover: Samil 100 6x6 artillery tractor or a 10 ton equivalent

ARMAMENT
Main Armament:
Caliber, Type, Name: 155-mm, canon
Barrel Length (cal): 45
Rate of Fire (rpm):
- Burst: 3
- Normal: 2
- Sustained: 2

Loader Type: Semi-automatic
Breech Type: Interrupted screw
Muzzle Brake Type: Single baffle
Traverse: (°):
- Left: 41
- Right: 41
- Total: 82

Elevation (°) (-/+): -3/+75°

FIRE CONTROL
Indirect Fire: Digital Panoramic Telescope
Direct Fire: Trunnion mounted telescopic sight
Collimator: INA
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
G-5 MkIII Upgrade of G-5 (see NOTES)

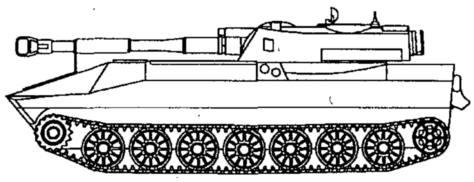
MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
155-mm Frag-HE, M1 HE
- Indirect Fire Range (m):
  - Minimum Range: 3000
  - Maximum Range: 30,000
- Complete Projectile Weight (kg): 8.7
- Muzzle Velocity: 897 m/s
- Fuze Type: PD M841

155-mm Frag-HE BB, M1 HE
- Indirect Fire Range (m):
  - Minimum Range: INA
  - Maximum Range: 39,000
- Complete Projectile Weight (kg): 8.7
- Muzzle Velocity: 895 m/s
- Fuze Type: PD M841

Other Ammunition Types: See NOTES

NOTES
The G5 is fully compatible with NATO standard 155-mm ammunition and has a direct fire range of 3000 meters (using a Frag-HE round). The APU, combined with the tandem walking-beam suspension, gives the G5 excellent self-propelled mobility over short distances. The four wheels are all powered and give the gun excellent traction over most terrain. But, the APU serves purposes other than mobility. It provides power to open and close the trails, raise and lower the trail wheels, and raise and lower the firing platform. However, there is no power traverse or elevation. Although designed for an eight-man section, the South African Defense Force normally operates the G5 with a five-man section. However, the G5 can operate with minimum of two people when all of the powered systems are working. The G-5 MkIII includes 35 reliability modifications and performance improvements. The improvements include the addition of the AS2000 Gun Monitor, an improved braking system, bigger diameter and wider trail wheels (specifically designed for sand), and incorporation of the REUTECH ACV 58 Communications System.

Russian 122-mm Self-Propelled Howitzer 2S1



Weapons & Ammunition Types	Typical Combat Load
122-mm howitzer	**45**
Frag-HE	
HEAT-FS	
Smoke	
Illumination	

SYSTEM
Alternative Designations: 122-mm 2S1 Gvozdika
Date of Introduction: 1974
Proliferation: At least 12 countries
Description:
Crew: 4 (section of 6 with 2 in ammo carrier)
Platform (chassis): MT-LBu
Combat Weight (mt): 15.7
Chassis Length Overall (m): 7.26
Height Overall (m): 2.72
Width Overall (m): 2.85

Automotive Performance:
Engine Type: V-8, 300 hp, Diesel
Cruising Range (km): 500 km
Speed (km/h):
Max Road: 60
Max Off-Road: 30
Cross-Country: INA
Max Swim: 4.5
Fording Depths (m): Amphibious
Emplacement Time (min): 2
Displacement Time (min): 1

Radio: R-123M

Protection:
Armor, Turret (mm): 20
Armor Turret Top (mm): 10
Armor Hull (mm): 15
Self-Entrenching Blade: No
NBC Protection System: Yes
Smoke Equipment: No

ARMAMENT
Main Armament:
Caliber, Type, Name: 122-mm, canon, 2A31
Barrel Length (cal): 36
Rate of Fire (rpm):
Burst: 5
Normal: 4
Sustained: 1-2
Fire from Ground: INA
Loader Type: Semi-automatic
Breech Type: Horizontal sliding wedge

Muzzle Brake Type: Double baffle
Traverse: (°):
Left: 360
Right: 360
Total: 360
Elevation (°) (-/+): -3/+70°

FIRE CONTROL
Indirect Fire: PG-2 Panoramic Telescope (PANTEL)
Direct Fire: OP 5-37
Collimator: K-1
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
None

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
122-mm Frag-HE, OF-81
Indirect Fire Range (m):
Minimum Range: 1000
Maximum Range: 15,300
Complete Projectile Weight (kg): 21.76 (OF-56)
Muzzle Velocity: 680 m/s
Fuze Type: RGM-2 PD

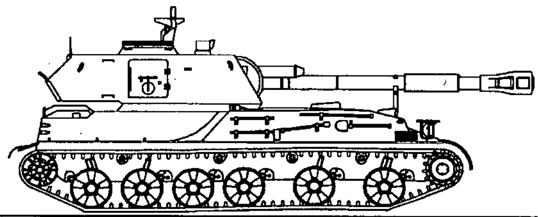
122-mm, HEAT-FS
Direct Fire Range (m):
Minimum Range: 0
Maximum Range: 1000
Armor Penetration (mm): 460 (@ 0° obliquity any range)
Complete Projectile Weight (kg): 21.58
Muzzle Velocity: 740 m/s
Fuze Type: GPV-2 PIBD

122-mm Frag-HE Rocket Assisted
Indirect Fire Range (m):
Minimum Range: INA
Maximum Range: 21,900
Complete Projectile Weight (kg): 21.76 (3OF-56)
Muzzle Velocity: INA
Fuze Type: PD

Other Ammunition Types: Incendiary, Chemical, Flechette, Expendable Jammer, Semi-active laser-guided Kitolov-2M Frag-HE

NOTES
The 2S1's ammunition stowage rack is not mechanized. The 2S1 is manually loaded with a semiautomatic ramming capability. The four-man crew consists of the commander, driver, gunner, and loader.

Russian 152-mm Self-Propelled Gun-Howitzer 2S3M



Weapons & Ammunition Types	Typical Combat Load
152-mm howitzer	**46**
Frag-HE Smoke Illumination	
7.62 PKT MG	**1500**

SYSTEM
Alternative Designations: 152-mm 2S3M Akatsiya
Date of Introduction: 1973
Proliferation: At least 8 countries
Description:
Crew: 4
Platform (chassis): Modified SA-4 Ganef
Combat Weight (mt): 27.5
Chassis Length Overall (m): 7.75
Height Overall (m): 3.13
Width Overall (m): 3.21

Automotive Performance:
Engine Type: 520-hpV-59 V-12 multi-fuel diesel
Cruising Range (km): 450 km
Speed (km/h):
Max Road: 60
Max Off-Road: 25
Cross-Country: INA
Max Swim: N/A
Fording Depth (m): 1.00
Emplacement Time (min): 3
Displacement Time (min): 3

Radio: R-123M

Protection:
Armor, Turret (mm): 20
Armor Turret Top (mm): 15
Armor Hull (mm): INA
Self-Entrenching Blade: Yes
NBC Protection System: Yes
Smoke Equipment: No

ARMAMENT
Main Armament:
Caliber, Type, Name: 152-mm, 2A33
Barrel Length (cal): 34
Rate of Fire (rpm):
Burst: 4
Normal: 3
Sustained: 1
Fire from Ground: INA
Loader Type: Semiautomatic
Breech Type: Vertical sliding wedge
Muzzle Brake Type: Double baffle
Traverse: (°):
Left: 360
Right: 360
Total: 360
Elevation (°) (-/+): -4/+60°

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm machinegun PKT
Mount Type: Bow (ball-mounted)
Direct Fire Range (m): 1000
Max Effective Range (m):
Day: 1000 /400-500 on the move
Night: 800
Fire on Move: Yes
Rate of Fire (rpm): 650 (cyclic)

FIRE CONTROL
Indirect Fire: PG-4 Panoramic Telescope (PANTEL)
Direct Fire: OP 5-38
Collimator: K-1
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
2S3M1: Upgrade of 2S3M

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
152-mm Frag-HE, OF32
Indirect Fire Range (m):
Minimum Range: 4600
Maximum Range: 17,400
Complete Projectile Weight (kg): 43.56 (OF25)
Muzzle Velocity: 655 m/s
Fuze Type: V-90 PD

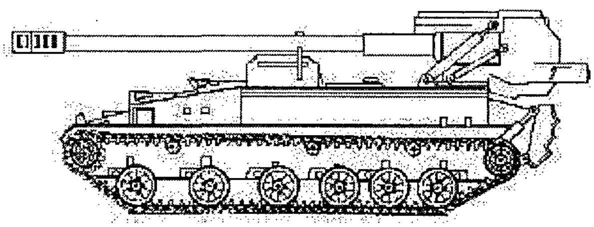
152-mm, HEAT, BP-540
Direct Fire Range (m):
Minimum Range: 0
Maximum Range: 1000
Armor Penetration (mm): INA
Complete Projectile Weight (kg): 27.00
Muzzle Velocity: 655 m/s
Fuze Type: GPV-3 PD

152-mm Frag-HE, OF-96
Indirect Fire Range (m):
Minimum Range: INA
Maximum Range: 24,400
Complete Projectile Weight (kg): 43.56 (OF-64)
Muzzle Velocity: INA
Fuze Type: PD

Other Ammunition Types: DPICM, DPICM-BB, Incendiary, Chemical, Flechette, Semi-active laser-guided Krasnopol-M Frag-HE

NOTES
The 2S3M is an upgrade version of the 2S3. The 2S3M turret contains the 2A33 cannon, fire-control equipment, ammunition storage space, and work positions for commander, gunner, and loader. The cannon extends beyond the vehicle front and has an electrical loader/rammer attached to the cradle. Ammunition is stored in the rear of the chassis and can be replenished through a hatch in the rear panel.

Russian 152-mm Self-Propelled Gun 2S5



Weapons & Ammunition Types	Typical Combat Load
152-mm howitzer	**30**
Frag-HE	
Smoke	
Illumination	
7.62 PKT MG	**1500**

SYSTEM
Alternative Designations: 152-mm 2S5 Giatsint-S
Date of Introduction: 1981
Proliferation: At least 4 countries
Description:
Crew: 5 (section of 7 with 2 in ammo carrier)
Platform (chassis): Modified 2S3
Combat Weight (mt): 28.2
Chassis Length Overall (m): 8.33
Height Overall (m): 2.76
Width Overall (m): 3.25

Automotive Performance:
Engine Type: 520-hp V-59-V-12 multi-fuel diesel
Cruising Range (km): 500 km
Speed (km/h):
Max Road: 63
Max Off-Road: 25
Cross-Country: INA
Max Swim: 4.5
Fording Depths (m): 1.05
Emplacement Time (min): 2
Displacement Time (min): 1

Radio: R-123M

Protection:
Armor, Turret (mm): INA
Armor Turret Top (mm): 15
Armor Hull (mm): 15
Self-Entrenching Blade: Yes
NBC Protection System: Yes
Smoke Equipment: None

ARMAMENT
Main Armament:
Caliber, Type, Name: 152-mm, canon, 2A37
Barrel Length (cal): 52
Rate of Fire (rpm):
Burst: 6
Normal: 5
Sustained: 1-2
Fire from Ground: INA
Loader Type: Semi-automatic
Breech Type: Horizontal sliding wedge
Muzzle Brake Type: Multi baffle
Traverse: (°):
Left: 15
Right: 15
Total: 30
Elevation (°) (-/+): -2/+57°

Auxiliary Weapon:
Caliber, Type, Name: 7.62-mm machinegun PKT
Mount Type: Bow (ball-mounted)
Direct Fire Range (m): 1000
Max Effective Range (m):
Day: 1000 /400-500 on the move
Night: 800
Fire on Move: Yes
Rate of Fire (rpm): 650 (cyclic)

FIRE CONTROL
Indirect Fire: PG-1M Panoramic Telescope (PANTEL)
Direct Fire: N/A
Collimator: K-1
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
None

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
152-mm Frag-HE, OF-39
Indirect Fire Range (m):
Minimum Range: 9100
Maximum Range: 28,400
Complete Projectile Weight (kg): 43.51 (OF-29)
Muzzle Velocity: 945 m/s
Fuze Type: V-429 PD

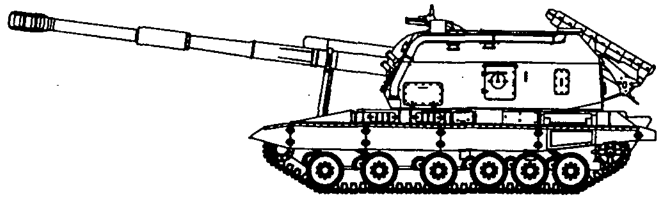
152-mm, HEAT, BP-540
Direct Fire Range (m):
Minimum Range: 0
Maximum Range: 1000
Armor Penetration (mm): INA
Complete Projectile Weight (kg): 27.00
Muzzle Velocity: 655 m/s
Fuze Type: GPV-3 PD

152-mm Frag-HE, OF-86
Indirect Fire Range (m):
Minimum Range: INA
Maximum Range: 30,500
Complete Projectile Weight (kg): 43.8 (OF-59)
Muzzle Velocity: 945 m/s
Fuze Type: V-429 PD

Other Ammunition Types: DPICM, DPICM-BB, Incendiary, Chemical, Flechette, Semi-active laser-guided Krasnopol-M Frag-HE

NOTES
The 2S5 is more powerful, has a longer range and a higher rate of fire than the 2S3. However, the 2S5 has a limited main armament traverse and a narrower elevation range than the 2S3.

Russian 152-mm Self-Propelled Howitzer 2S19



Weapons & Ammunition Types	Typical Combat Load
152-mm howitzer	**50**
Frag-HE Smoke Illumination	
12.7-mm MG	**300**

SYSTEM
Alternative Designations: 152-mm 2S19 Msta-S
Date of Introduction: 1989
Proliferation: At least 4 countries
Description:
Crew: 5 (section of 7 with 2 in ammo carrier)
Platform (chassis): Modified T-72
Combat Weight (mt): 42
Chassis Length Overall (m): 11.91
Height Overall (m): 2.98
Width Overall (m): 3.58

Automotive Performance:
Engine Type: 840-hp V84-A diesel
Cruising Range (km): 500 km
Speed (km/h):
Max Road: 60
Max Off-Road: 25
Cross-Country: INA
Max Swim: N/A
Fording Depths (m): Unprepared: 1.5
Emplacement Time (min): 1-2
Displacement Time (min): 1-2

Radio: R-173

Protection:
Armor, Turret (mm): 15
Armor Turret Top (mm): 15
Armor Hull (mm): 15
Self-Entrenching Blade: Capable of digging a complete firing pit in 40-60 minutes
NBC Protection System: Yes
Smoke Equipment: Six Type 902 smoke grenade launchers and Vehicle engine exhaust smoke system (VEESS)

ARMAMENT
Main Armament:
Caliber, Type, Name: 152-mm, canon, 2A64
Barrel Length (cal): 48
Rate of Fire (rpm):
Burst: 8
Normal: 6
Sustained: 2
Fire from Ground: 6-7
Loader Type: autoloader
Breech Type: Vertical sliding wedge
Muzzle Brake Type: Double baffle
Traverse: (°):
Left: 360
Right: 360
Total: 360
Elevation (°) (-/+): -4/+68°

Auxiliary Weapon:
Caliber, Type, Name: 12.7-mm NSVT machinegun
Mount Type: PZU-5 AA
Direct Fire Range (m): 2000
Max Effective Range (m):
Day: 1500 (AA)/1500 (Ground)
Night: N/A
Fire on Move: Yes
Rate of Fire (rpm): 800 (cyclic)

FIRE CONTROL
Indirect Fire: 1P22 Panoramic Telescope (PANTEL)
Direct Fire: 1P23
Collimator: K-1
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
None

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
152-mm Frag-HE, OF-72
Indirect Fire Range (m):
Minimum Range: 6500
Maximum Range: 24,700
Complete Projectile Weight (kg): 43.56 (OF-45)
Muzzle Velocity: 864 m/s
Fuze Type: RGM-2 PD

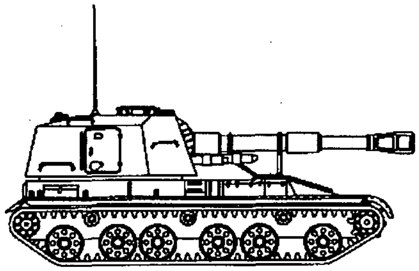
152-mm, HEAT, BP-540
Direct Fire Range (m):
Minimum Range: 0
Maximum Range: 1000
Armor Penetration (mm): INA
Complete Projectile Weight (kg): 27.00
Muzzle Velocity: 655 m/s
Fuze Type: GPV-3 PD

152-mm Frag-HE BB, OF-91
Indirect Fire Range (m):
Minimum Range: 6710
Maximum Range: 29,000
Complete Projectile Weight (kg): 42.86 (OF-61)
Muzzle Velocity: 828 m/s
Fuze Type: KZ-88 PD

Other Ammunition Types: All standard 152-mm artillery rounds

NOTES
The 2S19's gun crew can load the gun at any angle of elevation. The 2S19 can also produce a smokescreen by injecting diesel fuel into the exhaust outlet. The 21-hp gas turbine AP-18D Auxiliary Power Unit provides power for turret operations when the vehicle engine is shut down.

Chinese 152-mm Self-Propelled Gun-Howitzer Type 83



Weapons & Ammunition Types	Typical Combat Load
152-mm howitzer	30
Frag-HE Smoke Illumination	
12.7-mm MG	650
7.62-mm MG	650

SYSTEM
Alternative Designations: None
Date of Introduction: 1984
Proliferation: At least 1 country
Description:
Crew: 5
Platform (chassis): Type 83
Combat Weight (mt): 30.0
Chassis Length Overall (m): 7.33
Height Overall (m): 3.50
Width Overall (m): 3.24

Automotive Performance:
Engine Type: Type 12150L, V-12, 520-hp liquid-cooled diesel
Cruising Range (km): 450 km
Speed (km/h):
Max Road: 55
Max Off-Road: 35
Cross-Country: INA
Max Swim: N/A
Fording Depth (m): 1.3
Emplacement Time (min): 1
Displacement Time (min): 1

Radio: Type 889D

Protection:
Armor, Turret (mm): INA
Armor Turret Top (mm): INA
Armor Hull (mm): INA
Self-Entrenching Blade: No
NBC Protection System: No
Smoke Equipment: No

ARMAMENT
Main Armament:
Caliber, Type, Name: 152-mm, Type 66 cannon
Barrel Length (cal): 29
Rate of Fire (rpm):
Burst: 4
Normal: INA
Sustained: INA
Fire from Ground: INA
Loader Type: Semiautomatic
Breech Type: Vertical sliding wedge
Muzzle Brake Type: Double baffle
Traverse: (°):
Left: 360
Right: 360
Total: 360
Elevation (°) (-/+): -5/+65°

Auxiliary Weapon:
Caliber, Type, Name: 12.7-mm (12.7x108) AA MG Type 54
Mount Type: Turret top
Direct Fire Range (m): 1500
Max Effective Range (m):
Day: 1500 ground/1600 for air targets (APDS)
Night: INA
Fire on Move: Yes
Rate of Fire (rpm): 80-100 practical, 600 for air targets in 2-10 round bursts

Caliber, Type, Name: 7.62 (7.62 x 54R) Machinegun Type 59
Mount Type: Turret coax
Direct Fire Range (m): 1800
Max Effective Range (m):
Day: 1000
Night: 800
Fire on Move: Yes
Rate of Fire (rpm): 250 practical, 600 cyclic in 2-10 round bursts

FIRE CONTROL
Indirect Fire: Panoramic
Direct Fire: INA
Collimator: INA
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
425-mm Mineclearing Rocket Launcher Type 462: 2-round rocket launcher for use in clearing minefields.

120-mm SP Anti-Tank Gun: The AT gun is fitted with a 120-mm smoothbore mounted inside a turret on a Type 83 Gun-Howitzer chassis.

Chinese 152-mm Self-Propelled Gun-Howitzer Type 83 continued

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
152-mm Frag-HE, Type 66
- Indirect Fire Range (m):
  - Minimum Range: 9600
  - Maximum Range: 17,230
- Complete Projectile Weight (kg): 43.6
- Muzzle Velocity: 655 m/s
- Fuze Type: Liu-4 PD and Proximity

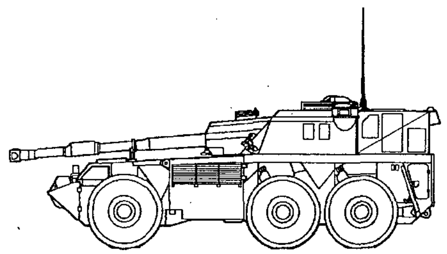
152-mm Frag-HE Rocket Assisted Projectile
- Indirect Fire Range (m):
  - Minimum Range: INA
  - Maximum Range: 21,880
- Complete Projectile Weight (kg): INA
- Muzzle Velocity: INA
- Fuze Type: PD

152-mm Frag-HE Type 83
- Indirect Fire Range (m):
  - Minimum Range: INA
  - Maximum Range: 30,370
- Complete Projectile Weight (kg): 46.95
- Muzzle Velocity: 955 m/s
- Fuze Type: Liu-4 PD and Proximity

Other Ammunition Types: HE-I, Illumination, Smoke

NOTES
The Type 83 152-mm SP Gun-Howitzer is capable of firing all standard types of 152-mm rounds. The main armament cannon is based on the Chinese 152-mm Towed Type 66 mounted on a vehicle hull similar to the Russian 152-mm SP Gun-Howitzer 2S3. The crew communicates with each other using the Type 803 intercom system. There are reports of the Type 83 being equipped with an anti-tank rocket launcher referred to as the Type 40. However, it is suspected that the rocket launcher is really the 40-mm anti-tank rocket launcher Type 69-1 (an upgraded variant of the Russian RPG-7).

South African 155-mm Self-Propelled Howitzer G6



Weapons & Ammunition Types	Typical Combat Load
155-mm howitzer	**45**
Frag-HE Smoke Illumination	
.50 Cal. M2 HB MG	**900**

SYSTEM
Alternative Designations: 155-mm G6 Rhino
Date of Introduction: 1988
Proliferation: At least 2 countries
Description:
Crew: 6
Platform (chassis): Purpose built 6x6 wheeled
Combat Weight (mt): 48
Chassis Length Overall (m): 10.4
Height Overall (m): 3.5
Width Overall (m): 3.4

Automotive Performance:
Engine Type: 525-hp air-cooled diesel
Cruising Range (km): 700 km
Speed (km/h):
Max Road: 85
Max Off-Road: 30
Cross-Country: INA
Max Swim: N/A
Fording Depth (m): 1.00
Emplacement Time (min): 1
Displacement Time (min): 0.5

Radio: INA

Protection:
Armor, Turret (mm): See NOTES
Armor Turret Top (mm): See NOTES
Armor Hull (mm): See NOTES
Self-Entrenching Blade: No
NBC Protection System: Yes
Smoke Equipment: 8 81-mm grenade launchers

ARMAMENT
Main Armament:
Caliber, Type, Name: 155-mm, canon
Barrel Length (cal): 45
Rate of Fire (rpm):
Burst: 3
Normal: 2
Sustained: 1
Fire from Ground: INA
Loader Type: Semi-automatic
Breech Type: Interrupted screw
Muzzle Brake Type: Single baffle
Traverse: (°):
Left: 40
Right: 40
Total: 80
Elevation (°) (-/+): -5/+75°

Auxiliary Weapon:
Caliber, Type, Name: .50 (12.7x99) heavy machinegun, M2HB
Mount Type: Cupola AA mount
Direct Fire Range (m): INA
Max Effective Range (m):
Day: 1000
Night: INA
Fire on Move: Yes
Rate of Fire (rpm): 450-550 (cyclic)

FIRE CONTROL
Indirect Fire: Digital Panoramic Telescope
Direct Fire: Trunnion mounted telescopic sight
Collimator: INA
Gun Display Unit: None
Fire Control Computer: None

VARIANTS
None

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
155-mm Frag-HE, M1 HE
Indirect Fire Range (m):
Minimum Range: 3000
Maximum Range: 30,000
Complete Projectile Weight (kg): 8.7
Muzzle Velocity: 897 m/s
Fuze Type: PD M841

155-mm Frag-HE BB, M1 HE
Indirect Fire Range (m):
Minimum Range: INA
Maximum Range: 39,000
Complete Projectile Weight (kg): 8.7
Muzzle Velocity: 895 m/s
Fuze Type: PD M841

Other Ammunition Types: See NOTES

South African 155-mm Self-Propelled Howitzer G6 continued

NOTES

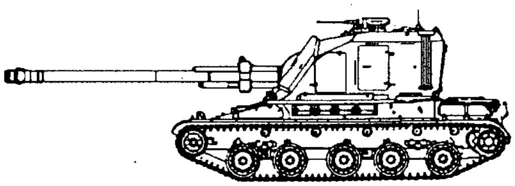
The G6 is a three-axle, six-wheeled, heavily armored system mounting a modified version of the G5 cannon. The G6 is fully compatible with NATO standard 155-mm ammunition and has a direct fire range of 3000 meters (using a Frag-HE round). The rigid chassis is actually divided into two parts, a driver's/engine compartment and a crew compartment. In order to distribute its weight and to maintain mobility over sand and soft terrain, the G6 employs large 21x25 run-flat tires. The driver controls a central tire-inflation system to vary the ground pressure. The system can also be used to maintain some degree of tire pressure in case of air leakage from small punctures. The G6 is equipped with an electronically controlled hydraulic flick rammer that provides an initial rate of fire of 3 rounds per minute.

The vehicle hull and turret provide protection against 7.62-mm small arms fire and artillery shrapnel. The frontal 60° arc provides protection against 20-mm type ammunition. Additionally, the shape and armor thickness of the chassis hull allows it to withstand at least three mine detonations (against TM46 antitank landmine or equivalent) before being immobilized. The separation of the driver/engine compartment from the crew compartment also facilitates survival against mines. The connection between the two is perforated with blowout holes to direct the force of the blast upwards, away from any personnel compartments. The separation also allows the driver to be beyond the detonation point before the mine is activated. The driver also has bullet-resistant glass windows that can be further protected by armored shutters, although it limits him to the use of a periscopic viewing port. The vehicle commander has limited steering and braking capability if the driver becomes a casualty. The crew compartment has four firing ports (two each side) so the crew can engage targets without exposing themselves to return fire.

A 45-hp (34 kw) Auxiliary Power Unit (APU) provides power for turret operations, recharging the batteries, and the driver/crew compartment air conditioning system. A wide range of optional subsystems is available to increase the efficiency of the G6 and its crew. They include the following:

- Inertial navigation and laying or back-up laying systems
- Night vision equipment
- Barrel cooling and thermal warning systems
- Fire control computer interface
- Muzzle velocity analyzer
- Explosion control for fuel tanks

French 155-mm Self-Propelled Howitzer AU-F1



Weapons & Ammunition Types	Typical Combat Load
155-mm howitzer	**42**
Frag-HE Smoke Illumination	
.50 Cal. M2 HB MG	**800**

SYSTEM
Alternative Designations: 155-mm GCT (Export Version)
Date of Introduction: 1979
Proliferation: At least 4 countries
Description:
Crew: 4
Platform (chassis): Modified AMX-30
Combat Weight (mt): 42.0
Chassis Length Overall (m): 10.25
Height Overall (m): 3.25
Width Overall (m): 3.15

Automotive Performance:
Engine Type: Hispano-Suiza HS110, 720-hp water-cooled multi-fuel
Cruising Range (km): 450 km
Speed (km/h):
Max Road: 60
Max Off-Road: 40
Cross-Country: INA
Max Swim: N/A
Fording Depth (m): 2.10
Emplacement Time (min): 1-2
Displacement Time (min): 1

Radio: TRC 559 (VHF-FM)

Protection:
Armor, Turret (mm): See NOTES
Armor Turret Top (mm): See NOTES
Armor Hull (mm): See NOTES
Self-Entrenching Blade: No
NBC Protection System: Yes
Smoke Equipment: 4 grenade launchers

ARMAMENT
Main Armament:
Caliber, Type, Name: 155-mm, canon
Barrel Length (cal): 40
Rate of Fire (rpm):
Burst: 8
Normal: 6
Sustained: 2-3 (manual loading)
Fire from Ground: INA
Loader Type: Autoloader
Breech Type: Vertical sliding wedge
Muzzle Brake Type: Double baffle
Traverse: (°):
Left: 360
Right: 360
Total: 360
Elevation (°) (-/+): -4/+66°

Auxiliary Weapon:
Caliber, Type, Name: .50 (12.7x99) heavy machinegun, M2HB
Mount Type: Cupola AA mount
Direct Fire Range (m): INA
Max Effective Range (m):
Day: 1000
Night: INA
Fire on Move: Yes
Rate of Fire (rpm): 450-550 (cyclic)

FIRE CONTROL
Indirect Fire: M 589 Optical Gonimeter
Direct Fire: INA
Collimator: INA
Gun Display Unit: ATILA fire direction system
Fire Control Computer: None

VARIANTS
AU-F1T: Ugrade of AU-F1

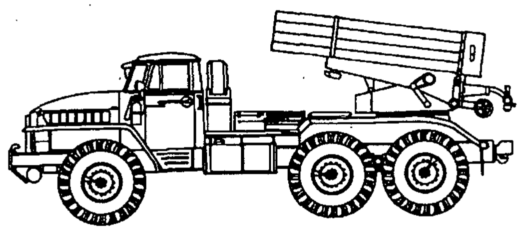
MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
155-mm Frag-HE, OE-155-56/69
Indirect Fire Range (m):
Minimum Range: 9600
Maximum Range: 23,000
Complete Projectile Weight (kg): 43.75
Muzzle Velocity: 810 m/s
Fuze Type: PD

155-mm Frag-HE Rocket Assisted H3
Indirect Fire Range (m):
Minimum Range: INA
Maximum Range: 31,500
Complete Projectile Weight (kg): INA
Muzzle Velocity: 830 m/s
Fuze Type: PD

Other Ammunition Types: DPICM, Illumination, Smoke

NOTES
The export version of the AU-F1 is known as the GCT (Grande Cadence de Tir or high rate of fire). The AU-F1T is fitted with the Sagem Cita 20 inertial navigation system as well as a 20-24 hp gas turbine auxiliary power unit (APU). A four-man gun crew can reload the AU-F1 in 15 minutes. A two-man gun crew can reload the AU-F1 in 20 minutes. The AU-F1's armor provides crew protection against artillery shrapnel and small arms fire.

Russian 122-mm Multiple Rocket Launcher BM-21



Weapons & Ammunition Types	Typical Combat Load
122-mm rocket	
Frag-HE	40

SYSTEM
Alternative Designations: BM-21 GRAD (Hail) MRL
Date of Introduction: 1963
Proliferation: At least 50 countries
Description:
Crew: 5 (8 with 9K51 Complex)
Chassis/Carriage: Ural 375-D 6x6 wheeled
Combat Weight (mt): 13.7
Chassis Length Overall (m): 7.35
Height Overall (m): 3.09
Width Overall (m): 2.40

Automotive Performance:
Engine Type: ZIL 375, 180 hp water-cooled, V-8 gasoline engine
Cruising Range (km): 450 km
Speed (km/h):
- Max Road: 75
- Max Off-Road: 35
- Cross-Country: INA
- Max Swim: N/A

Fording Depths (m): Unprepared: 1.5
Emplacement Time (min): 3
Displacement Time (min): 2

Radio: R-123M

Protection:
Armor, Front (mm): None
Armor Side (mm): None
Armor Roof (mm): None
Self-Entrenching Blade: No
NBC Protection System: No
Smoke Equipment: No

ARMAMENT
Launcher:
Caliber, Type, Name: 122-mm, 9P132
Number of Tubes: 40 (4 rows of 10 tubes)
Launch Rate:
- Full Salvo Time: 40 rounds in 20 seconds
- Single Rocket Interval: .5 seconds per rocket

Loader Type: Manual
Reload Time: 10 minutes
Launcher Drive: Electric
Traverse: (°):
- Left: 102
- Right: 70
- Total: 172

Elevation (°) (-/+): - 0/+55°

FIRE CONTROL
Indirect Fire: PG-1M Panoramic Telescope (PANTEL)
Collimator: K-1
Fire Control Computer: None
Position Location System: None

VARIANTS
BM-21V: Russian 12-tube version for airborne divisions
BM-21B: Russian 36-tube MRL on a 6x6 ZIL-131 chassis
Grad-P: Russian 1 round rocket launcher
BM-11: North Korean 30-tube version
Type 81: Chinese 40- rail-launched version
RM-70: Czechoslovakian 40-tube version
Sakr: Egyptian 40- tube version

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
122-mm Frag-HE, 9M22U
Indirect Fire Range (m):
- Minimum Range: 5000
- Maximum Range: 20,380

Warhead Weight (kg): 18.4 (M21OF)
Rocket Length: (m): 2.87
Maximum Velocity: INA
Fuze Type: MRV-U (PD)

122-mm Frag-HE, 9M28F
Indirect Fire Range (m):
- Minimum Range: 1500
- Maximum Range: 15,000

Warhead Weight (kg): 21.0
Rocket Length: (m): 2.87
Maximum Velocity: INA
Fuze Type: MRV-U (PD) or AR-6 (proximity)

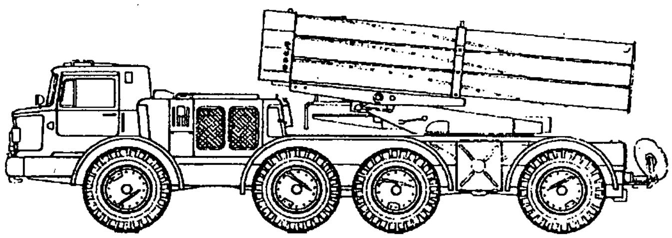
122-mm Frag-HE, Type 90A (Chinese)
Indirect Fire Range (m):
- Minimum Range: 12,700
- Maximum Range: 32,700

Warhead Weight (kg): 18.3
Rocket Length: (m): 2.75
Maximum Velocity: INA
Fuze Type: PD

Other Ammunition Types: Smoke, Incendiary, Chemical, RF Jammer, Illumination, Antitank mines, Antipersonnel mines

NOTES
The BM-21 is unquestionably the world's most widely used MRL. The launcher with supporting equipment is referred to as the complex 9K51. A special electric generator powers the launcher. The 9V170 firing device is cab mounted. But, the rockets can be fired using a remote-firing device that has a 64-meter-long cable.

Russian 220-mm Multiple Rocket Launcher 9P140



Weapons & Ammunition Types	Typical Combat Load
220-mm rocket	**16**
Frag-HE	

SYSTEM
Alternative Designations: 9P140 Uragan
Date of Introduction: 1977
Proliferation: At least 7 countries
Description:
Crew: 4
Chassis/Carriage: ZIL-135LM 8x8 wheeled
Combat Weight (mt): 20.0
Chassis Length Overall (m): 9.3
Height Overall (m): 3.2
Width Overall (m): 2.8

Automotive Performance:
Engine Type: 2 each - 177 hp, 8 cylinder, 4-stroke gasoline engines
Cruising Range (km): 500 km
Speed (km/h):
Max Road: 65
Max Off-Road: INA
Cross-Country: INA
Max Swim: N/A
Fording Depths (m): Unprepared: 1.2
Emplacement Time (min): 3
Displacement Time (min): 3

Radio: R-123M

Protection:
Armor, Front (mm): None
Armor Side (mm): None
Armor Roof (mm): None
Self-Entrenching Blade: No
NBC Protection System: No
Smoke Equipment: No

ARMAMENT
Launcher:
Caliber, Type, Name: 220-mm, 9P140
Number of Tubes: 16 (2 rows of 6 tubes and 1 row of 4 tubes)
Launch Rate:
Full Salvo Time: 16 rounds in 20 seconds
Single Rocket Interval: 1.25 seconds per rocket
Loader Type: Manual
Reload Time: 15-20 minutes
Launcher Drive: Electric
Traverse: (°):
Left: 30
Right: 30
Total: 60
Elevation (°) (-/+): -0/+55°

FIRE CONTROL
Indirect Fire: PG-1M Panoramic Telescope (PANTEL)
Collimator: K-1
Fire Control Computer: None
Position Location System: None

VARIANTS
None

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
220-mm Frag-HE, 9M27F
Indirect Fire Range (m):
Minimum Range: 10,000
Maximum Range: 35,000
Warhead Weight (kg): 100
Rocket Length: (m): 4.8
Maximum Velocity: INA
Fuze Type: Electronic timing (ET)

220-mm DPICM, 9M27K
Indirect Fire Range (m):
Minimum Range: 10,000
Maximum Range: 35,000
Warhead Weight (kg): 90
Rocket Length: (m): 5.1
Maximum Velocity: INA
Fuze Type: Electronic timing (ET)

220-mm Antitank, 9M27K2
Indirect Fire Range (m):
Minimum Range: 10,000
Maximum Range: 35,000
Warhead Weight (kg): 90
Rocket Length: (m): 5.1
Maximum Velocity: INA
Fuze Type: Electronic timing (ET)

Russian 220-mm Multiple Rocket Launcher 9P140 continued

MAIN ARMAMENT AMMUNITION (continued) **Caliber, Type, Name:** 220-mm Antipersonnel, 9M27K3 Indirect Fire Range (m): Minimum Range: 10,000 Maximum Range: 35,000 Warhead Weight (kg): 90 Rocket Length: (m): 5.1 Maximum Velocity: INA Fuze Type: Electronic timing (ET)	220-mm Antitank, 9M59 Indirect Fire Range (m): Minimum Range: 10,000 Maximum Range: 35,000 Warhead Weight (kg): 90 Rocket Length: (m): 5.1 Maximum Velocity: INA Fuze Type: Electronic timing (ET) **Other Ammunition Types:** None

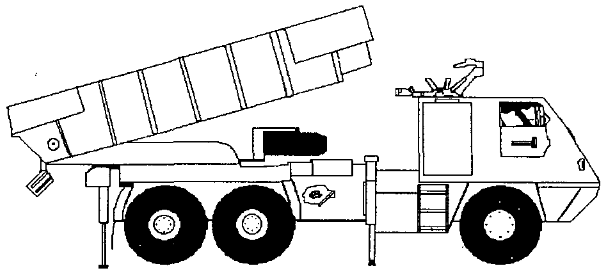
NOTES

The 9P140 Uragan (previously referred to incorrectly as BM-22 or BM-27) is the world's first modern fin and spin-stabilized heavy rocket system. Essentially a scaled-up version of the BM-21, the 9P140 use many of the same design features. The launcher, 9T452 transloader, rockets, and support equipment constitutes the 9K57 complex.

The 9P140 and its transloader are both based on variants of the gasoline-powered ZIL-135LM 8-ton 8x8 chassis. The truck is unusual in that it uses two engines, each driving the wheels on one side of the truck, and only the front and rear axles steer. The 9P140 cab has a blast shield that is raised during firing, and the vehicle is stabilized during firing by two manually emplaced hydraulic jacks at the rear of the chassis.

The launcher has electrically powered traversing and elevating mechanisms. During travel, the launcher assembly is oriented rearward and a light sheet metal cover over the muzzle end of the tubes prevents foreign material from entering the tube. This is a safety feature that is designed for travel when loaded. There is no such cover for the muzzle end of an unloaded launcher.

Brazilian 127-mm, 180-mm, & 300-mm Multiple Rocket Launcher ASTROS II



Weapons & Ammunition Types	Typical Combat Load
127-mm rocket Frag-HE	32
180-mm rocket Frag-HE	16
300-mm rocket Frag-HE	4
.50 Cal. M2 HB MG	INA

SYSTEM
Alternative Designations: ASTROS II AV-LMU
Date of Introduction: 1983
Proliferation: At least 6 countries
Description:
Crew: 3
Chassis/Carriage: TECTRAN 10-ton 6x6 wheeled
Combat Weight (mt): 20.0
Chassis Length Overall (m): 8.0
Height Overall (m): 2.6
Width Overall (m): 2.4

Automotive Performance:
Engine Type: 280 hp, water-cooled turbocharged, diesel engine
Cruising Range (km): INA
Speed (km/h):
Max Road: 70
Max Off-Road: 40
Cross-Country: INA
Max Swim: N/A
Fording Depths (m): Unprepared: 1.0
Emplacement Time (min): INA
Displacement Time (min): INA

Radio: INA

Protection:
Armor, Front (mm): None
Armor Side (mm): None
Armor Roof (mm): None
Self-Entrenching Blade: No
NBC Protection System: No
Smoke Equipment: 6 smoke grenade launchers

ARMAMENT
Launcher:
Caliber, Type, Name: 127-mm, 180-mm, 300-mm, ASTROS
Number of Tubes: 127-mm (32), 180-mm (16), 300-mm (4)
Launch Rate:
Full Salvo Time: INA
Single Rocket Interval: INA
Loader Type: Manual
Reload Time: INA
Launcher Drive: Electric
Traverse: (°):
Left: INA
Right: INA
Total: INA
Elevation (°) (-/+): INA

Auxiliary Weapon:
Caliber, Type, Name: .50 (12.7x99) heavy machinegun, M2HB
Mount Type: Cab AA mount
Direct Fire Range (m): INA
Max Effective Range (m):
Day: 1000
Night: INA
Fire on Move: Yes
Rate of Fire (rpm): 450-550 (cyclic)

FIRE CONTROL
Indirect Fire: INA
Collimator: INA
Fire Control Computer: FIELDGAURD Radar or the FILA System
Position Location System: INA

VARIANTS:
None

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
127-mm Frag-HE, SS-30
Indirect Fire Range (m):
Minimum Range: 9000
Maximum Range: 30,000
Warhead Weight (kg): INA
Rocket Length: (m): 3.9
Maximum Velocity: INA
Fuze Type: INA

Other Ammunition Types: None

Caliber, Type, Name:
180-mm Frag-HE, SS-40
Indirect Fire Range (m):
Minimum Range: 15,000
Maximum Range: 35,000
Warhead Weight (kg): INA
Rocket Length: (m): 4.2
Maximum Velocity: INA
Fuze Type: INA

Other Ammunition Types: DPICM, HE-Incendiary, Antitank mines, Antipersonnel mines, Runway Denial

Brazilian 127-mm, 180-mm, & 300-mm Multiple Rocket Launcher ASTROS II continued

Caliber, Type, Name: 300-mm Frag-HE, SS-60 Indirect Fire Range (m): Minimum Range: 20,000 Maximum Range: 60,000 Warhead Weight (kg): INA Rocket Length: (m): 5.6 Maximum Velocity: INA Fuze Type: INA **Other Ammunition Types:** DPICM, HE-Incendiary, Antitank mines, Antipersonnel mines, Runway Denial	300-mm Frag-HE, SS-80 Indirect Fire Range (m): Minimum Range: 22,000 Maximum Range: 90,000 Warhead Weight (kg): INA Rocket Length: (m): 5.6 Maximum Velocity: INA Fuze Type: INA **Other Ammunition Types:** DPICM, HE-Incendiary, Antitank mines, Antipersonnel mines, Runway Denial

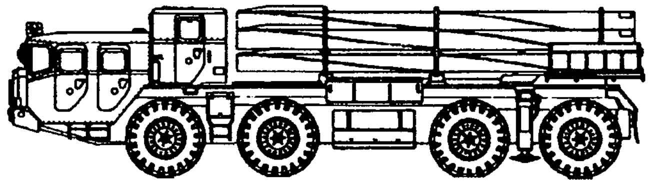
NOTES

The ASTROS (Artillery SaTuration ROcket System) II is a modular multiple rocket launcher capable of firing three different caliber wrap-around fin rockets (for improved accuracy) using several types of warheads. The universal modules enable the system to accomplish fire missions with ranges from 9 to 90 kilometers.

The ASTROS II system consists of the following vehicles:
Universal Multiple Launcher (AV-LMU), Ammunition Supply Vehicle (AV-RMD), Command and Control Vehicle/Fire Control Unit (AV-VCC), Mobile Workshops (for field maintenance), and the Optional Electronic Fire Control Unit (AV-UCF). All of the ASTROS II vehicles use the Tectran Enginharia 10 ton, 6x6, wheeled vehicle chassis.

A typical firing battery consists of six AV-LMU launchers, six AV-RMD ammunition supply vehicles, and one AV-VCC fire control unit. A AV-VCC command and control unit and two mobile workshops are found at battalion level. The battalion level AV-VCC can coordinate and direct fire missions for three ASTROS batteries. The AV-RMD ammunition supply vehicle carries two complete loads for each launcher.

Russian 300-mm Multiple Rocket Launcher 9A52-2



Weapons & Ammunition Types	Typical Combat Load
300-mm rocket	12
Frag-HE	

SYSTEM
Alternative Designations: 9A52-2 Smerch-M
Date of Introduction: 1989
Proliferation: At least 4 countries
Description:
Crew: 4 (7 with 9K58 Complex)
Chassis/Carriage: MAZ-543M 8x8 wheeled
Combat Weight (mt): 43.7
Chassis Length Overall (m): 12.1
Height Overall (m): 3.05
Width Overall (m): 3.05

Automotive Performance:
Engine Type: 518 hp, V-12 diesel engine
Cruising Range (km): 850 km
Speed (km/h):
- Max Road: 60
- Max Off-Road: 35
- Cross-Country: INA
- Max Swim: N/A

Fording Depths (m): Unprepared: 1.1
Emplacement Time (min): 3
Displacement Time (min): 3

Radio: R-123M

Protection:
Armor, Front (mm): None
Armor Side (mm): None
Armor Roof (mm): None
Self-Entrenching Blade: No
NBC Protection System: No
Smoke Equipment: No

ARMAMENT
Launcher:
Caliber, Type, Name: 300-mm, 9A52
Number of Tubes: 12 (3 rows of 4 tubes)
Launch Rate:
- Full Salvo Time: 12 rounds in 38 seconds
- Single Rocket Interval: 3 seconds per rocket

Loader Type: Manual
Reload Time: 36 minutes
Launcher Drive: Electric
Traverse: (°):
- Left: 30
- Right: 30
- Total: 60

Elevation (°) (-/+): -0/+55°

FIRE CONTROL
Indirect Fire: PG-1M Panoramic Telescope (PANTEL)
Collimator: K-1
Fire Control Computer: None
Position Location System: None

VARIANTS
None

MAIN ARMAMENT AMMUNITION
Caliber, Type, Name:
300-mm Frag-HE, 9M55F
- Indirect Fire Range (m):
  - Minimum Range: 20,000
  - Maximum Range: 70,000
- Warhead Weight (kg): 258
- Rocket Length: (m): 7.6
- Maximum Velocity: INA
- Fuze Type: Electronic timing (ET)

300-mm DPICM, 9M55K
- Indirect Fire Range (m):
  - Minimum Range: 20,000
  - Maximum Range: 70,000
- Warhead Weight (kg): 235
- Rocket Length: (m): 7.6
- Maximum Velocity: INA
- Fuze Type: Electronic timing (ET)

300-mm Sensor-fuzed (MOTIV-3M), 9M55K1
- Indirect Fire Range (m):
  - Minimum Range: 20,000
  - Maximum Range: 70,000
- Warhead Weight (kg): 233
- Rocket Length: (m): 7.6
- Maximum Velocity: INA
- Fuze Type: Electronic timing (ET)

Other Ammunition Types: Smoke, Incendiary, Chemical, Leaflet, Fuel Air Explosive (FAE), R-90 expendable miniature UAV (experimental)

NOTES
The 9A52-2 launcher with all supporting equipment, including the 9T234-2 Transloader, and the 1K123 Vivary Fire Control System, is referred to as the complex 9K58.

Chapter 7
Air Defense

This chapter provides an overview of selected air defense systems either in use or readily available to an OPFOR. The selection of weapons is not intended to be all-inclusive, but rather a representative sampling of weapons and equipment supporting various OPFOR military capabilities.

This chapter is divided into three categories—***towed AA guns***, ***self-propelled AA guns/combination guns*** and ***surface-to-air missiles (SAMs)***. ***Towed AA guns*** covers, in order, the KS-19M2 100-mm gun, S-60 57-mm gun and the ZU-23 23-mm gun. The next category, ***self-propelled AA guns/combination guns***, contains the ZSU-23-4 23-mm gun and the 2S6 30-mm gun/missile system. The final category of ***surface-to-air missiles (SAMs)*** consists of the SA-7b, SA-8b, SA-14, SA-15b and the SA-18.

Tactical air defense is used to protect ground force units and other potential targets from attack by enemy fixed-wing aircraft and armed helicopters. Due to increases in performance and the sheer number of air defense systems, specifically manportable systems, the selected systems represent some of the most formidable threats to aircraft of all types.

Some trends in air defense development will become more widespread in the near future. These include the production of authorized and unauthorized copies of existing systems and the development of hybrid systems. The sensor package may consist of one or more radars, direct view optics, and electro-optics systems. The sensor package is the single most important aspect of air defense systems since these devices perform the surveillance and tracking functions. As the data classification permits, all attempts have been made to provide the user with as much information as possible in these areas. Radar systems have traditionally been the most popular sensor for air-defense systems, however, with the latest generation weapons they are usually supplemented with a variety of optic or electro-optic sensors such as; TV cameras, night vision sights, and laser rangefinders. As the trends become more defined and more information becomes available, updates to the systems will be produced.

Questions and comments on data listed in this chapter should be addressed to:

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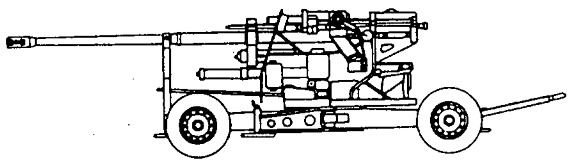
Russian 100-mm Towed AA Gun KS-19M2



Weapons & Ammunition Types	Typical Combat Load
100-mm gun	100
Frag-HE	
AP-T	
APC-T	

SYSTEM
Alternative Designations: None
Date of Introduction: 1949
Proliferation: At least 20 countries

Description:
Crew: 15
Carriage: Towed 2-axle, 4-wheel carriage
Combat Weight (kg): 11,000
Length Overall (m): 9.3
  Travel Position: 9.45
  Firing Position: INA
Length of Barrel (m): 5.74
Height (m):
  Overall: 2.2
  Travel Position: INA
  Firing Position: 7.62
Width Overall (m): 2.32
Prime Mover: Towing vehicle AT-S or AT-T

Automotive Performance:
Max. Towed Speed (km/h): 35
Emplacement Time (min): 7
Displacement Time (min): 6

ARMAMENT
Gun:
Caliber, Type: 100-mm gun
Number of Barrels: 1
Service Life of Barrel (rds): 2,800
Rate of Fire(rd/min):
  Maximum: INA
  Practical: 10-15
Loader Type: Manual
Reload Time (min): INA
Traverse (°): 360
Traverse Rate (°/sec): 20
Elevation (°) (-/+): -3 to 89
Elevation Rate (°/sec): 12
Reaction time (sec): 30

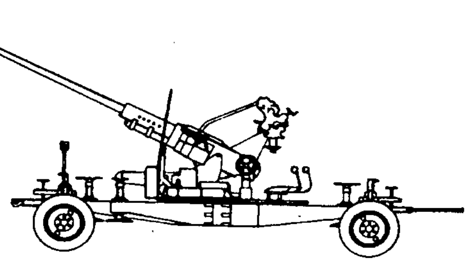
FIRE CONTROL
On-carriage:
PO-1M telescope
  Field of View (°): 14
  Power: 5x
PG panoramic telescope:
  Field of View (°): 10
  Power: 4x
Off-carriage:
Rangefinder: D-49 (off carriage)
Radar:
  Name: SON-9/SON-9A (FIRE CAN)
  Function: Fire Control
  Detection Range (km): 80
  Tracking Range (km): 35
  Frequency: 2.7-2.9 GHz
  Frequency Band: E
  Peak Power (kW): 300
PUAZO 6-19 or 6-19M fire control director

VARIANTS
Type 59: Chinese variant.

MAIN ARMAMENT AMMUNITION
Types: Frag-HE, AP-T, APC-T
Range (m):
  With on-carriage sight: 4,000
  With off-carriage radar: 12,600
Projectile Weight (kg):
  Frag-HE: 15.61
  AP-T: 15.89
  APC-T: 16
Muzzle Velocity (m/s): 900-1,000
Fuze Type: Proximity and Time
Self-Destruct (sec): 30

NOTES
The KS-19M2 may also be employed in a ground support role.

Russian 57-mm Towed AA Gun S-60



	Weapons & Ammunition Types	Typical Combat Load
	57-mm gun	200
	FRAG-T APC-T	

SYSTEM
Alternative Designations: None
Date of Introduction: 1950
Proliferation: At least 46 countries

Description:
Crew: 7
Carriage: Four-wheel
Weight (kg): 4,500
Length Overall (m):
Travel Position: 8.50
Firing Position: 8.84
Length of Barrel (m): 4.39
Height (m):
Overall:
Travel Position: 2.37
Firing Position: 6.02
Width Overall (m):
Travel Position: 2.08
Firing Position: 6.9
Prime Mover: Ural-375D

Automotive Performance:
Max. Towed Speed (km/h): 60
Emplacement Time (min): 1
Displacement Time (min): 3

ARMAMENT
Gun:
Caliber, Type: 57-mm automatic cannon
Number of Barrels: 1 each
Service Life of Barrel (rds): INA
Rate of Fire (rd/min):
Cyclic: 105-120
Practical: 70
Loader Type: 4 rd clip, manual
Reload Time (sec): 4-8
Traverse (°): 360
Traverse Rate (°/sec): 40
Elevation (°) (-/+): -4 to +87
Elevation Rate (°/sec): 34
Reaction time (sec): 4.5

FIRE CONTROL
On-carriage:
Optical mechanical computing sight
AZP-57:
Target Range (m): 5,500
Direct fire telescope

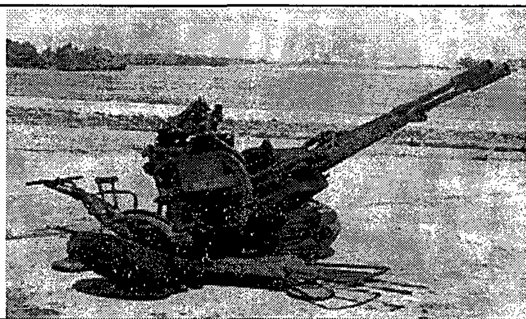
Off-carriage: (see NOTES)
Rangefinder: D-49
Radar:
Name: SON-9/SON-9A
Function: Fire Control
Detection Range (km): 80
Tracking Range (km): 35
Frequency: 2.7-2.9 GHz
Frequency Band: E
Peak Power (kW): 300
PUAZO 6-60 fire control director

VARIANTS
Type 59: Chinese variant
SZ-60: Hungarian license-built variant

MAIN ARMAMENT AMMUNITION
Type: FRAG-T, APC-T
Range (m):
With on-carriage sight: 4,000
With off-carriage radar: 6,000
Projectile Weight (kg):
FRAG-T: 2.81
APC-T: 2.82
Muzzle Velocity (m/s): 1,000
Fuze Type:
FRAG-T: Point detonating
APC-T: Base detonating
Self-Destruct (sec): 13-17

NOTES
Some versions may have the FLAP WHEEL as the primary fire control radar. A S-60 battery will generally consist of six guns, a fire-control radar, and a fire-control director. Four-round clips feed ammunition horizontally into weapon. The S-60 also has an ammunition ready rack that can hold 4 four-round clips near ammunition feed mechanism on left side of the breech. The S-60 can also be used in a ground support role.

Russian 23-mm Towed AA Gun ZU-23



Weapons & Ammunition Types	Typical Combat Load
2 x 23-mm AA guns	**2,400**
,HE-I HEI-T API-T TP	

SYSTEM
Alternative Designation: None
Date of Introduction: 1962
Proliferation: At least 50 countries

Description:
Crew: 5
Carriage: Two-wheeled
Combat Weight (kg): 950
Length Overall (m):
Travel Position: 4.57
Firing Position: 4.60
Length of Barrel (m): 2.01
Height (m):
Overall:
Travel Position: 1.87
Firing Position: 1.28
Width Overall (m):
Travel Position: 1.83
Firing Position: 2.41
Prime Movers: GAZ-69 4 x 4 truck, MTLB-T, BMD-2

Automotive Performance:
Max. Towed Speed (km/h): 70
Emplacement Time (sec): 15-20
Displacement Time (sec): 35-40

ARMAMENT
Gun:
Caliber, Type: 23-mm, gas-operated gun
Number of Barrels: 2
Breech Mechanism: Vertical Sliding Wedge
Rate of Fire (rd/min):
Cyclic: 800-1,000
Practical: 200
Feed: 50-rd ammunition canisters fitted on either side of the upper mount assembly
Loader Type: Magazine
Reload Time (sec): 15
Traverse (^{0}): 360
Traverse Rate (^{0}/sec): INA
Elevation (^{0}) (-/+): -10°to +90°
Elevation Rate: (^{0}/sec): 54
Reaction Time (min): 8 (est.)

FIRE CONTROL

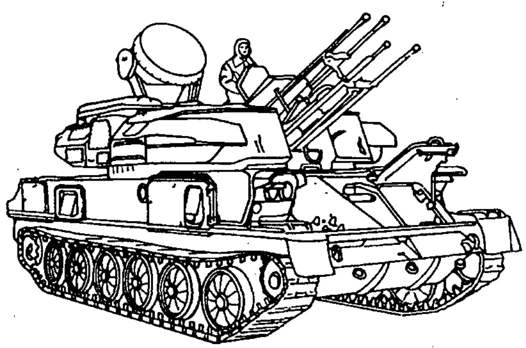
Sights w/magnification:
Optical mechanical sight for AA fire
Straight tube telescope for ground targets

VARIANTS
ZU-23M: Egyptian produced ZU-23, also referred to as the SH-23M.

MAIN ARMAMENT AMMUNITION
Type: HE-I, HEI-T, API-T, TP
Range (m):
Max. Range: 2,500
Min. Range: INA
Altitude (m):
Max. Altitude: 1,500
Min. Altitude: INA
Projectile Weight (kg):
HE-I: 0.18
HEI-T: 0.19
API-T: 0.189
TP: 0.18
Muzzle Velocity (m/s): 970
Fuze Type:
HE-I: Point detonating
HEI-T: Point detonating
API-T: Base detonating
TP: Dummy
Self-Destruct (sec): 8

NOTES
Highly mobile air dropable system. Fires the same ammunition as the ZSU-23-4. The reload time will depend on the proficiency of the crew to manually reload. Can fire from the traveling position in emergencies. The ZU-23 can also be used in a ground support role.

Russian 23-mm SP AA Gun ZSU-23-4



Weapons & Ammunition Types	Typical Combat Load
4x 23-mm AA guns	**2,000**
HE-I HEI-T API-T	

SYSTEM
Alternative Designation: Shilka
Date of Introduction: 1965
Proliferation: At least 28 countries

Description:
Crew: 4
Combat Weight (mt): 20.5
Chassis: GM-575 Tracked, six road wheels, no track support rollers
Length (m): 6.5
Height (m):
Radar up: 3.75
Radar down: 2.60
Width (m): 3.1

Automotive Performance:
Engine Type: V6R-1 diesel
Cruising Range (km): 450
Speed (km/h):
Max. Road: 50

Radio: R-123

Protection:
NBC Protection System: Yes

ARMAMENT
Gun:
Caliber, Type, Name: 23-mm liquid-cooled AA 2A7/2A7M
Rate of Fire(rd/min):
Practical: INA
Cyclic: 850-1,000
Reload Time (min): 20
Elevation (°) (-/+):-4° to +85°
Fire on Move: Yes
Reaction Time (sec): 12-18

FIRE CONTROL
Sights w/magnification:
Day and night vision devices:
Driver periscope: BMO-190
Driver IR periscope: INA
Commander periscope: TPKU-2
Commander IR periscope: TKH-ITC
IFF: INA

Radar: 1RL33M1
Name: GUN DISH
Function: Search and Tracking
Detection Range (km): 20
Tracking Range (km): 10
Frequency: 14.8 to 15.6 GHz
Frequency Band: J

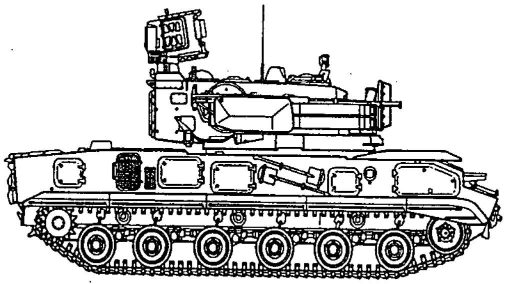
Optical-mechanical computing sight: Part of fire-control subsystem designated as RPK-2

VARIANTS
(see NOTES)

MAIN ARMAMENT AMMUNITION
Types: HE-I, HEI-T, API-T
Range (m):
Max. Range: 2,500
Min. Range: INA
Altitude (m):
Max. Altitude: 5,100 (about 3,500 w/self-destruct fuzing)
Min. Altitude: INA
Projectile Weight (kg):
HE-I: 0.18
HEI-T: 0.19
API-T: 0.189
Muzzle velocity (m/s): 950-1,000
Fuze Type:
HE-I: Point detonating
HEI-T: Point detonating
APT-T: Base detonating

NOTES
Ammunition is normally loaded with a ratio of three HE rounds to one AP round. ZSU 23-4 Shilka, is capable of acquiring, tracking and engaging low-flying aircraft (as well as mobile ground targets while either in place or on the move). Resupply vehicles carry an estimated additional 3,000 rounds for each of the four ZSUs in a typical battery. Recent (October 1997) information details ZSU-23-4 updates/modernization being offered by the Ukrainians that include: a new radar system replacing the GUN DISH radar, plus a sensor pod believed to include day/night camera, and a laser rangefinder; and mounted above radar/sensor pod is a layer of six fire-and-forget SAMs, believed to be Russian SA-18/GROUSE.

Russian 30-mm SP AA Gun/Missile System 2S6M



Weapons & Ammunition Types	Typical Combat Load
2 x 30-mm twin barrel cannons	**1,904**
AP-T	
Frag-T	
HE-I	
SA-19/GRISON	**8**

SYSTEM
Alternative Designations: 2K22M, Tunguska-M
Date of Introduction: 1990
Proliferation: At least 2 countries

Description:
Crew: 4
Combat Weight (mt): 34
Chassis: GM-352M tracked vehicle
Chassis Length Overall (m): 7.93
Height (m):
TAR up: 4.02
TAR down: 3.36
Width Overall (m): 3.24

Automotive Performance:
Engine Type: V-12 turbo diesel
Cruising Range (km): 500
Speed (km/h):
Max. Road: 65
Max. Swim: INA
Fording Depths (m): INA

Radio: R-173

Protection:
NBC Protection System: Yes

ARMAMENT
Gun:
Caliber, Type, Name: 30-mm gun, 2A38M
Rate of Fire (rd/min): 4,800 (four gun total)
Reload Time (min): gun ammunition and missiles in about 16 min.
Elevation (°) (-/+): -10 to + 87°
Fire on Move: Yes

Missile: 9M311
Name: SA-19/GRISON
Range (m):
Max. Range: 8,000-10,000 (see NOTES)
Min. Range: 2,500
Altitude (m):
Max. Altitude: 3,500
Min. Altitude: 15
Dimensions:
Length (m): 2.83
Weight (kg): 57 (in container)
Missile Speed (m/s): 600-900
Guidance: SACLOS
Seeker Field of View(°) : INA
Tracking Rate: INA
Warhead Type: Frag-HE
Warhead Weight (kg): 9
Fuze Type: Proximity
Self-Destruct (sec): INA
System Reaction Time (sec): 6-12
Fire on Move: No (must be at a halt to fire the missile)

FIRE CONTROL
Sights w/magnification:
Stabilized optical sight 1A29M
Magnification: 8x
Field of View(°): 8°
Commander's position IR day/night sight
IFF: Yes

Radars: HOT SHOT radar system
Name: 1RL144 (TAR)
Function: Target Acquisition
Detection Range (km): 18-20
Tracking Range (km): INA
Frequency: 2-3 GHz
Frequency Band: E

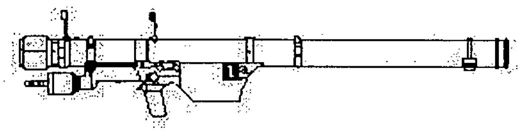
Name: 1RL144M (TTR)
Function: Target Tracking
Detection Range (km): 16
Tracking Range (km): INA
Frequency: 10-20 GHz
Frequency Band: J

VARIANTS
(see NOTES)

MAIN ARMAMENT AMMUNITION
Type: AP-T, Frag-T, HE-I
Range (m):
Max. Range: 4,000
Min. Range: 200
Altitude (m):
Max. Altitude: 3,000
Min. Altitude: 0
Projectile Weight (kg): INA

NOTES
Range out to 10 km for hovering aircraft and low flying targets. In addition to the 8 mounted ready missiles two additional missiles can be carried inside. There is a 2S6M1 variant/upgrade, which has improved missile control, range and altitude capabilities of 1.5-10 km, and 0.015-6 km respectively. However, as of November 1997 the 2S6M1 is not known to be fielded.

Russian Manportable SAM System SA-7b/GRAIL



Weapons & Ammunition Types	Typical Combat Load
ready missile	1

SYSTEM
Alternative Designation:
9K32M Strela-2M
Date of Introduction: 1972
Proliferation: Worldwide

Description:
Crew: 1

ARMAMENT
Launcher
Name: 9P54M
Dimensions:
Length (m): 1.47
Diameter (mm): 70
Weight (kg): 4.71
Reaction Time (acquisition to fire) (sec): 5-10
Time Between Launches (sec): INA
Reload Time (sec): 6-10

Missile
Name: 9M32M
Range (m):
Max. Range: 5,500
Min. Range: 500
Altitude (m):
Max. Altitude: 4,500
Min. Altitude: 18
Dimensions:
Length (m): 1.40
Diameter (mm): 70
Weight (kg): 9.97
Missile Speed (m/s): 580
Propulsion: Solid fuel booster and solid fuel sustainer rocket motor.
Guidance: Passive IR homing device (operating in the medium IR range)
Seeker Field of View(°): 1.9°
Tracking Rate(°/sec): 6°
Warhead Type: HE
Warhead Weight (kg): 1.15
Fuze Type: Contact (flush or grazing)
Self-Destruct (sec): 15

FIRE CONTROL
Sights w/Magnification:
Launcher has sighting device and a target acquisition indicator. The gunner visually identifies and acquires the target.
Gunner:
Field of View (°): INA
Acquisition Range (m): INA

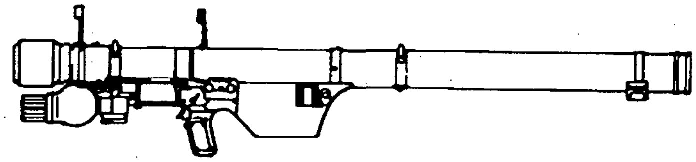
IFF: Yes (see NOTES)

VARIANTS
SA-N-5: Naval version
HN-5A: Chinese version
Strela 2M/A: Yugoslavian upgrade
Sakr Eye: Egyptian upgrade

Mounted in several types of vehicles in four, six, and eight-tube launcher varieties.
Can be mounted on several helicopters (Mi-24, S-342 Gazelle)

NOTES
The seeker is fitted with a filter to reduce the effectiveness of decoying flares and to block IR emissions. This missile is a tail-chasing heat (IR) seeker that depends on its ability to lock on to heat sources of usually low-flying fixed- and rotary-wing aircraft. An identification friend or foe (IFF) system can be fitted to the gunner/operator's helmet. Further, a supplementary early warning system consisting of a passive RF antenna and headphones can be used to provide early cue about the approach and rough direction of an enemy aircraft. The main difference between the SA-7 and SA-7b is the improved propulsion of the SA-7b. This improvement increases the speed and range of the newer version.

Russian Manportable SAM System SA-14/GREMLIN



Weapons & Ammunition Types	Typical Combat Load
ready missiles	1

SYSTEM
Alternative Designation: 9K34 Strela-3
Date of Introduction: 1978
Proliferation: Worldwide

Description:
Crew: 1

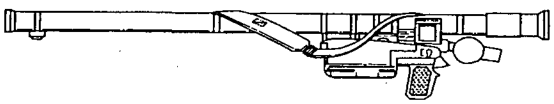
ARMAMENT
Launcher
Name: 9P59
Dimensions:
Length (m): 1.40
Diameter (mm): 75
Weight (kg): 2.95
Reaction Time (sec): 14
Time Between Launches (sec): 35-40
Reload Time (sec): 25

Missile
Name: 9M36 or 9M36-1
Range (m):
Max. Range: 6,000
Min. Range: 600
Altitude (m):
Max. Altitude: 6,000
Min. Altitude: 50
Dimensions:
Length (m): 1.4 m
Diameter (mm): 75 mm
Fin Span (mm): INA
Weight (kg): 10.3
Missile Speed (m/s): 600
Propulsion: 2-stage solid-propellant rocket
Guidance: passive IR homing
Seeker Field of View: INA
Tracking Rate: INA
Warhead Type: Frag-HE
Warhead Weight (kg): 1.0
Fuze Type: Contact/grazing
Self-Destruct (sec): 14-17

FIRE CONTROL
Sights w/Magnification:
Launch tube has simple sights
Gunner:
Field of View (°): INA
Acquisition Range (m): INA
IFF: Yes

VARIANTS
Igla 9M39 (SA-N-8): Naval version

Russian Manportable SAM System SA-18/GROUSE



Weapons & Ammunition Types	Typical Combat Load
ready missiles	1

SYSTEM
Alternative Designation: 9K38 Igla
Date of Introduction: 1983
Proliferation: At least 4 countries

Description:
Crew: 1

ARMAMENT
Launcher
Name: 9P39
Dimensions (m):
Length: 1.708
Diameter: INA
Weight (kg): 1.63
Reaction Time (sec): 6-7
Time Between Launches (sec): 16
Reload Time (sec): 10

Missile
Name: 9M39
Range (m):
Max. Range: 6,000
Min. Range: 500
Altitude (m):
Max. Altitude: 3,500
Min. Altitude: 10
Dimensions (mm):
Length: 1708
Diameter: 70
Weight (kg): 10.6
Missile Speed: Mach 2
Propulsion: Solid fuel booster and dual-thrust solid fuel sustainer rocket motor.
Guidance: Passive IR homing
Seeker Field of View: INA
Tracking Rate: INA
Warhead Type: HE
Warhead Weight (kg): 1.27
Fuze Type: Contact
Self-Destruct (sec): 15

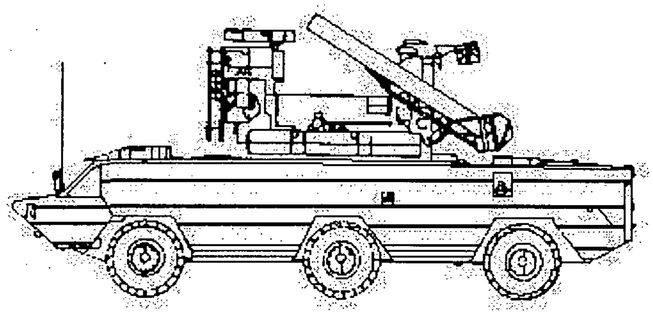
FIRE CONTROL
Sights w/Magnification:
Launcher has fore and rear sights
Gunner:
Field of View (°): INA
Acquisition Range (m): INA

IFF: Yes

VARIANTS
Igla-V: Air-to-air version
Igla-D: Use in airborne forces
Igla-N: Increased lethality
Igla-S: Improved version of Igla-N

NOTES
The SAM gunner is provided information about location and direction of approaching target(s) using a portable electronic plotting board. Two variants (Igla-D and Igla-N) can be separated in two parts for easier portability, but this adds 60 seconds to the reaction time. Igla-N is heavier due primarily to the warhead mass increased to 3.5 kg.

Russian SAM System SA-8b/GECKO



Weapons & Ammunition Types	Typical Combat Load
SA-8b in canisters	**6**

SYSTEM
Alternative Designations: 9K33M3 Osa-AKM
Date of Introduction: 1980
Proliferation: At least 25 countries

Description:
Crew: 3
Combat Weight (mt): 9
TELAR: BAZ-5937 6x6 amphibious cross-country capable vehicle
Length (m): 9.14
Height (m): 4.2 (with surveillance radar folded down)
Width (m): 2.75

Automotive Performance:
Engine Type: D20K300 diesel
Cruising Range (km): 500
Speed (km/h):
Max. Road: 80
Max. Swim: 8

Radio: R-123M

Protection:
NBC Protection System: Yes

ARMAMENT
Launcher:
Name: 9P35M2
Dimensions:
Length (m): 3.2
Diameter (mm): INA
Weight (kg): 35
Reaction Time (sec): INA
Time Between Launches (sec): 4
Reload Time (min): 5
Fire on Move: No
Emplacement Time (min): 4
Displacement Time (min): Less than 4 (est.)

Missile:
Name: 9M33M3
Range (m):
Max. Range: 15,000
Min. Range: 200
Altitude (m):
Max. Altitude: 12,000
Min. Altitude: 10
Dimensions (mm):
Length: 3158
Diameter: 209.6
Weight (kg): 170
Missile Speed (m/s): 1020
Propulsion: Solid propellant rocket motor
Guidance: RF CLOS
Warhead Type: Frag-HE
Fuze Type: Contact and proximity
Warhead Weight (kg): 16
Self-Destruct (sec): 25-28

FIRE CONTROL
Sights w/Magnification: INA
LLLTV/optical assist (for target tracking in low visibility and heavy ECM)

IFF: Yes

Radar:
Name: LAND ROLL
Function: Target Acquisition
Detection Range (km): 20-30
Tracking Range (km): 20-25
Frequency: 6-8 GHz
Frequency Band: H

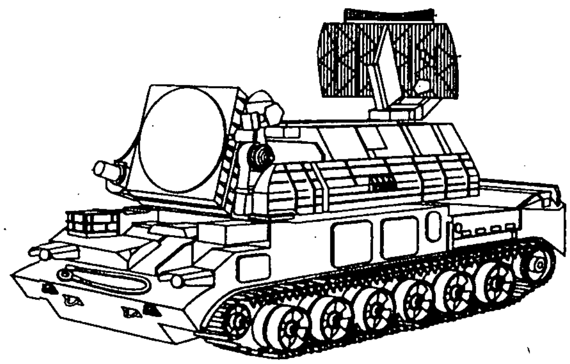
Radar:
Name: Monopulse Target Tracking Radar
Function: Target Tracking
Detection Range (km): 20-25
Tracking Range (km): INA
Frequency: 14.2-14.8 GHz
Frequency Band: J

2 Missile tracking radars:
Frequency: 10-20 GHz

VARIANTS
SA-8a: Initial production model that carries four missiles on exposed rails.
4K33 Osa-M (SA-N-4): Naval variant

NOTES
The first production version of this system was identified as SA-8a, which only had 4 launcher rails and exposed missiles. The SA-8b typically has two BAZ-5937 resupply/transloader vehicles, carrying 18 missiles each (boxed in sets of three) that supports a battery of four TELARs. A target can be brought under fire both with one missile as well as a volley of two missiles. This system is also air transportable.

Russian SAM System SA-15b/GAUNTLET



Weapons & Ammunition Types	Typical Combat Load
ready missiles	8

SYSTEM
Alternative Designations: 9K331 Tor-M1
Date of Introduction: 1990
Proliferation: At least 5 countries

Description:
Crew: 3
TLAR: 9A331 combat vehicle
Chassis: GM-355
Combat Weight (mt): 34
Length (m): 7.5
Height (m): 5.1 (TAR up)
Width (m): 3.3

Automotive Performance:
Engine Type: V-12 diesel
Cruising Range (km): 500
Speed (km/h):
Max. Road: 65

Radio: INA

Protection:
NBC Protection System: Yes

ARMAMENT
Launcher:
Name: INA
Dimensions: INA
Length (m): INA
Diameter (mm): INA
Weight (kg): INA
Reaction Time (sec): 5-8
Time Between Launches (sec): (see NOTE)
Reload Time (min): 10
Fire on Move: Yes
Emplacement Time (min): 5
Displacement Time (min): Less than 5

Missile:
Name: 9M331
Range (m):
Max. Range: 12,000
Min. Range: 100
Altitude (m):
Max. Altitude: 6,000
Min. Altitude: 10
Dimensions (mm):
Length: 2,900
Diameter: 235
Weight (kg): 167
Missile Speed (m/s): 850
Propulsion: INA
Guidance: Command
Warhead Type: Frag-HE
Fuze Type: RF Proximity
Warhead Weight (kg): 15
Self-Destruct (sec): INA

FIRE CONTROL
Sights w/Magnification:
Electro-optical (EO) television system:
Range: 20 km

IFF: Yes

Radar:
Name: INA
Function: Target Acquisition
Detection Range (km): 25
Tracking Range (km): INA
Frequency: INA
Frequency Band: H-band Doppler

Radar:
Name: INA
Function: Target Tracking and Guidance
Detection Range (km): INA
Tracking Range (km): 25
Frequency: INA
Frequency Band: K-band Doppler, Phased Array

VARIANTS
SA-N-9: Naval version

NOTES
SA-15b is designed to be a completely autonomous air defense system (at division level), capable of surveillance, command and control, missile launch and guidance functions from a single vehicle. The basic combat formation is the firing battery consisting of four TLARs and the Rangir battery command post. The TLAR carries eight ready missiles stored in two containers holding four missiles each. The SA-15b has the capability to automatically track and destroy 2 targets simultaneously in any weather and at any time of the day.

Chapter 8
Engineer and Logistics

This chapter provides the basic characteristics of selected ***engineer equipment*** and ***logistics vehicles***. ***Engineer equipment*** covers, in order, obstacle- and route-clearing vehicles, mine-laying systems, and mineclearing systems. It does not include engineer equipment designed primarily for civil engineering or construction in the rear areas. Also not included is dredging and gap crossing equipment. Data sheets addressing some of these systems will be sent with the next supplement to this guide.

The second category—***logistics vehicles***, provides the basic characteristics of selected trucks readily available to the OPFOR. It includes a representative vehicle from the light, utility, medium, and heavy truck categories. Later updates of this guide will include data on a wider selection of trucks, trailers, vans and other logistical equipment.

Questions and comments on data listed in this chapter should be addressed to:

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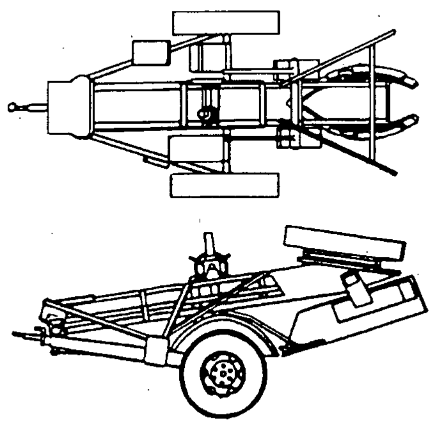
Russian Towed Mechanical Minelayer PMR-3



Mine Types	Typical Combat Load
Mines TM-44 TM-46 TM-57 TM-62 Series TM-72 TMD-B	(varies, see **Prime Mover**)

SYSTEM

Alternative Designations: INA
Date Of Introduction: INA
Proliferation: At least 17 countries

Description:
Crew: 6 (commander, driver, four operators)
Weight (mt): 1.3
Length (m): 5.6
Height (m): 2.7
Width (m): 2
Prime Mover:
6x6 ZIL-131 truck (200 mines) or
4x4 URAL-375D (350 mines) or
BTR-152 (120 mines)

MINELAYING EQUIPMENT

Operating Speed (km/h):
Burying: 2 to 3
Surface Laying: 4 to 10
In Snow: INA
Minelaying Rate (min): 10 to 12
Minelaying Pattern: Straight line
Mine Spacing (m): 3 to 4
Mine Capacity: Prime-mover dependent
Max Burial Depth (cm): 20

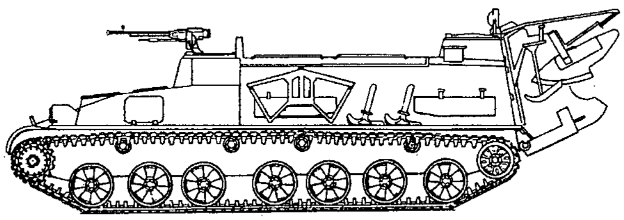
VARIANTS

PMZ-4: Lays controlled minefields; uses the same mines as the PMR-3 with the exception of the controllable minefield and cable-laying options; uses the UMP-2 Controlled AT Minefield Set

NOTES

The PMR-3, shown above, (and the similar PMZ-4) consists of a single chute and a plow attachment. Although both systems look similar at first glance, there are significant differences. Most notably, is the addition of a cable layer on the PMZ-4, used for the laying controlled minefields and the absence of the conveyer-belt chain drive on the wheels. Additionally, the PMZ-4 is more automated and must be hand loaded only. The towed-minelayers are used in sections of three or four and operate 20 to 40 meters apart with each minelayer laying a straight-line row. The mines in different rows are staggered with the distance between mines depending on whether the mines are pressure-initiated or full-width attack (influenced or tiltrod fuzed).

Russian Tracked Minelaying Vehicle GMZ-3



Mine Types	Typical Combat Load
Mines	**208**
TM-57 w/fuze MVZ-57	
TM-62 series w/fuzes	
TM-46	
TMD-B	
MV4-62	
MVP-62 & w/prox fuze	
MVN-80	
7.62-mm PKT MG	**3,000**

SYSTEM

Alternative Designations: INA
Date of Introduction: GMZ series-1963
Proliferation: Former Soviet Union

Description:
Crew: 3 (see NOTES)
Chassis: Based on the SA-4 (GANEF) SAM
Weight (mt): 28.5
Length (m): 8.62
Height (m): 2.7
Width (m): 3.25
Ground Clearance (mm): 470
Gradient (°): 30
Fording Depth (m): 1
Vertical Step (m): .7

AUTOMOTIVE

Engine: 4 cyl, 513 hp, muli-fuel diesel
Cruising Range (km): 500
Speed (km/h):
On Road: 60
Off Road: 30
Fuel Capacity (liters): INA
Night Driving Equipment: Yes, TVNE-4B for the driver and K-3A for the vehicle commander (and PKT)
Navigation Equipment: (see NOTES)
Radio: R-123
NBC Protection System: Yes
Smoke Screening System: VEESS, plus 6 81-mm launchers, 3 on each side.

MINELAYING EQUIPMENT

Operating Speed (km/h):
Burying: 6
Surface Laying: 16
In Snow: 10
Minelaying Pattern: Straight line or staggered
Mine Spacing (m): 5 and 10
Burial Depth (mm):
Ground: 120
Snow: 500
Length of Single-row Minefield (m):
Percussion Fuzes: 1,000
Proximity Fuzes: 2,000
Mine Capacity: 208
Mine Weight (kg): up to 12
Time Required to Load Minelayer with One Basic Mine Load (min) (7 men): 15 to 20
Men required to Load Minelayer with Mines: 7 (squad)
Time Required to Load Minelayer with Crew Only (m): 60
Time from Travel to Operating Position (min):
Automatic: Up to 2
Manual: Up to 8

ARMAMENT
Some GMZ may be armed with either the 12.7 or the 14.5 machineguns.

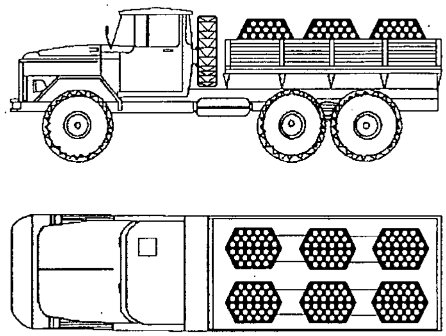
Main Armament:
Caliber, Type, Name: 7.62-mm PKT MG
Mount Type: Cupola (GMZ-3)
Max Effective Range (m):
Day: 2,000
Night: INA
Fire on Move: Yes
Rate of Fire (rd/min):
Practical: 250
Cyclic: 650

VARIANTS

GMZ: (shown above)
GMZ-2: (see NOTES)

NOTES
The crew of the GMZ-3 consists of three people—the vehicle commander, driver-mechanic, and the minelayer operator. The commander and driver are located in the forward section while the operator compartment is located in the rear portion of the vehicle. The vehicle commander operates the 7.62-mm PKT machinegun. The GMZ-3 has a digital navigation system allowing precise topographic tie-in of the minefield being laid. The previous model minelayer (GMZ-2) was not designed for the employment of mines with proximity fuzes.

Russian Scatterable Minelaying System UMZ

	Mine Types	Typical Combat Load
	**Mines** PFM-1 PFM-1/S POM-1 (S) POM-2S PTM-1S (PGMDM) PTM-3	(varies with type of mine—see below)

SYSTEM

Alternative Designations: Multipurpose Minelayer
Date of Introduction: INA
Proliferation: Former Soviet Union

Description:
Crew: 2 (driver and operator)
Chassis: ZIL-131 Truck (see VARIANTS)
Weight (mt):
Without Mine Load: 8.3
With Mine Load: 10
Length (m): 7.1
Height (m): 2.5
Width (m): 3
Ground Clearance (mm): 330

AUTOMOTIVE

Engine: V8, 150 hp, gas
Cooling: Water
Cruising Range (km): 525
Speed (km/h): 80
Gradient (°): 30
Fording Depth (m): 1.4
Vertical Step (m): .53
Night Vision Equipment: Yes, PNV-57E
Navigation Equipment: INA
Radio: R-159

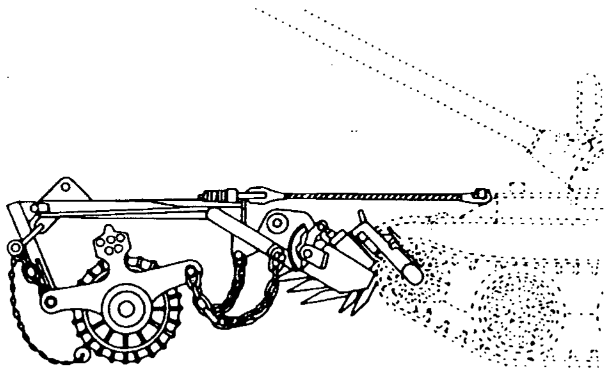
MINELAYING EQUIPMENT

Operating Speed (km/h): 10 to 40
Distance Mines Launched from Vehicle (m): 30-60
Minefield (m):
Length: 1,000 to 1,200
Depth: 30 to 120
Max Length of Minefield with One Basic Load (m):
AP, PFM-1S: 3,200
AP, POM-2: 5,000
AT, PTM-3: 600
Length of Triple-Row Minefield (m): 150 to 1,500 depending on mine type
Mine Capacity: From 180 to 11,520 depending on the type of mine
Number of Mines in One Basic Load:
AP, PFM-1S: 11,520
AP, POM-2: 720
AT, PTM-3: 180
Time Required to Load Minelayer (hr) (2 men): 1.5 to 2
Time from Travel to Operating Position (min): 5

VARIANTS
Although primarily mounted on the ZIL-131, the UMZ minelaying system has been observed mounted on several different carriers such as a modified MTLB-U chassis or on a PT-S tracked amphibious personnel carrier.

NOTES
While the UMZ, scatterable, mine system has been disclosed as the likely replacement for the GMZ-series, mechanical mineplanters, it probably will supplement the role formerly held by the GMZ. The UMZ consists of three launchers mounted on each side of the vehicle for a total of six mine launchers per vehicle. Each full turn launcher is hexagonally shaped and contains 30 launch tubes totaling 180. It can fire the mines to one or both sides, or to the rear. Both AP and AT mines are launched from the 140-mm launch tubes. The UMZ uses the same mine canisters as the PKM system. Depending on the position of the launch tubes, one-, two-, or three-lane mine fields can be laid.

Russian Tank-Mounted Mineclearing Roller-Plow KMT-5



SYSTEM	MINECLEARING EQUIPMENT
Alternative Designations: INA **Date of Introduction:** 1960s **Proliferation:** At least 20 countries **Description:** Weight Total (kg): 7,500 Roller Section: 2,265 Plough: 420 Length (m): 3.2 Width (m): 4 Ditch Crossing (m): 2.5 **System Components:** Two plows (KMT-4) and two sets of three rollers	**Type:** Roller and plow **Platform:** Mounted on T-54, T-55, T-62, other medium tanks **Form:** 3 rollers x 2 **Number of Rollers Per Set:** 2 **Total Number of Rollers:** 6 **Mine Removal Speed (km/h):** 8-12 **Cleared Lane Width, each (mm):** 810 x 2 **Mineclearing Track Width (mm):** 810 **Installation Time (min):** 30 to 45

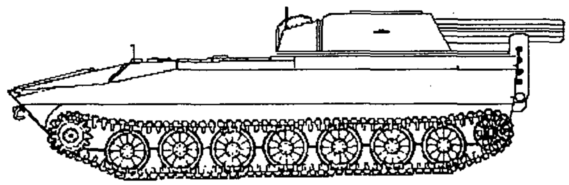
NOTES

The KMT-5M mine roller-plow is very flexible, since it allows for either the plows or the rollers to be used. The rollers function satisfactorily against mines equipped with simple pressure fuzes, but other mines will defeat this equipment. However, the roller-plow combination also allows the tank to counter more sophisticated fuzes with plows designed to uncover or push mines aside. The plows and rollers cannot work simultaneously.

The KMT-5M also includes a luminous lane-marking device for night operations. Because plows and rollers do not clear the area between them a "dogbone" or light chain with rollers is stretched between the roller sections to defeat tilt-rod mines. Quick disconnects allow the operator to drop either plows or rollers or both; otherwise, the crew can remove the system in 8 to 13 minutes. All current medium tanks have fittings for attaching mineclearing equipment.

There is one plow per tank platoon and one roller per company. For tanks newer than the T-55/62 the plows are no longer carried in the engineer company, but are permanently mounted on the tank. Therefore the engineers need only to transport the rollers. One KrAZ-255B truck (with KM-61 crane) or two ZIL-131 trucks can carry one KMT-5M.

Russian Tracked Mineclearing Vehicle MTK-2



SYSTEM

Alternative Designations: UR-77 mineclearing vehicle, M1979
Date of Introduction: 1981
Proliferation: FSU and former Warsaw Pact armies

Description:
Crew: 2 (commander-operator, driver-mechanic)
Chassis: Based on the 2S1
Weight (mt): 15.5
Length (m): 8.4
Height (m): 3.1
Width (m): 2.8
System Components: Vehicle and two mineclearing charges

AUTOMOTIVE

Cruising Range (km): 500
Speed (km/h):
On Road: 60
Off Road: 30
Water: 5
NBC Protection System: Yes
Smoke Screening System: No

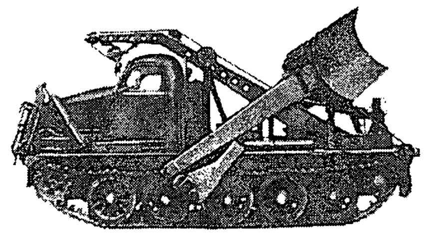
MINECLEARING EQUIPMENT

Type: Explosive line
Charges Used: UZP-77, UZ-67
Length of Charge (m): 93
Length of Charge Feed (m):
UZP-77: 200 and 500
UZ-67: 200 and 350
Size of Lane in AT Minefield (m):
Width: Up to 6
Length (USP-77): 80-90
Length (UZ-67): 75-80
Breaching Time (min): 3 to 5

VARIANTS (INA)

NOTES
The MTK-2 clears lanes in minefields by using rocket propelled charges. The charges are launched onto the minefield and then detonated by the vehicle commander-operator from within the vehicle. The charge can be fired on land or in the water.

Russian Tracked Route-Clearing Vehicle BAT-M



SYSTEM

Alternative Designations: Dozer
Date of Introduction: 1967
Proliferation: Widespread

Description:
Crew: 2
Chassis: AT-T heavy tracked artillery tractor
Weight (mt): 26
Length Overall (m): 10
Height Travel (m): 3.5
Width Overall (m): 4.7
Clearance (mm): 425
Gradient (°): 30
Trench Crossing (m): 1.57
Fording Depth (m): .7
Vertical Step (m): 1
Time from Travel to Operating Position (min): 5 to 7

AUTOMOTIVE

Engine: V12, 415 hp, diesel
Cruising Range (km): 500
Speed (km/h): 35
Navigation Equipment: No
NBC Protection: Yes
Radio: INA

BLADE

Width (m): 4.8
Blade Rate (m^3/hr): 250
Operating Speed (km/h): 10

ROTARY CRANE

Capacity (mt): 2

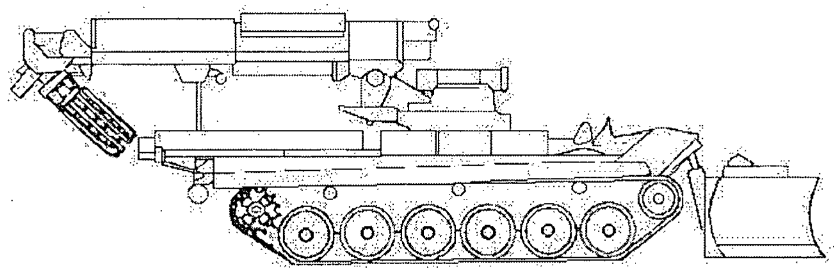
VARIANTS

BAT
BAT-2: Based on MT-T artillery tractor

NOTES

The BAT tractor dozer is a AT-T heavy tractor with a large dozer blade mounted at the front of the hull. It is designed for general engineer use, road and trail clearing and construction. The BAT-M is an improved model (over the BAT) and is electrohydraulic, whereas the BAT is electropneumatic. The BAT-M also has a hydraulic crane, and the dozer blade can be swung to the rear improving the vehicle's load distribution when in travelling mode.

Russian Obstacle Clearing Vehicle IMR-2M



SYSTEM
Alternative Designations: N/A
Date of Introduction: 1982
Proliferation: FSU and former Warsaw Pact armies

Description:
Crew: 2
Chassis: T-72A
Weight (mt): 44.3
Length (traveling) (m): 9.55
Height (traveling) (m): 3.68
Width (traveling) (m): 3.73
Gradient (°): 25
Fording Depth (m): 1.2
System Components: Multipurpose dozer equipment, boom, treadway mine exploder

AUTOMOTIVE

Engine: 12 cyl, 840 hp, diesel
Cruising Range (km): 500
Speed (km/h): 60
Night Driving Equipment: Yes
Radio: R-173 radio, R-174 intercom
NBC Protection System: Yes
Smoke Screening System: INA

BLADE
Can be used as a dozer, grader and V-blade, vertical plane skew ability.

Operating Speed (bulldozer) (km/h): 8-12
Earth Displacement (m^3/hr): 300
Lane Clearing Rate (km): .35

ALL-PURPOSE TOOL:

Trench Digging (1.1 to 1.3m deep)(m^3/hr): 8-10
Pit Digging (up to 2.5m deep) (m^3/hr): 12-16

BOOM

Capacity (mt): 2
Reach (m): 8.4

MINE SWEEPING SPEED (km):

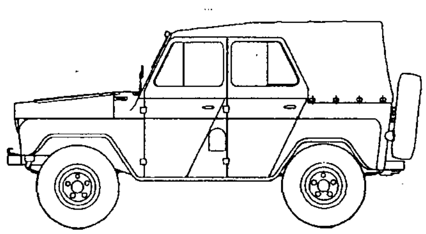
AT pressure mines: 6-15
Tilt Rod mines: 7

VARIANTS

IMR: The IMR is a NBC-protected, combat engineer vehicle based on the T-54/55 tank chassis. It is fitted with an articulating dozer blade and a telescoping crane that fits a number of attachments.
IMR-2: Equipped with mine sweepers and mine-clearing extended charges. Line drawing is of IMR-2.

NOTES
The IMR-2M differs from the IMR-2 in that the IMR-2M has no line-launched mineclearing charge. The IMR-2M has more armor, hydraulic equipment and a scraper-ripper.

Russian 0.6 mt 4 x 4 Utility Truck UAZ-469



SYSTEM

Alternative Designations: INA
Date of Introduction: 1973
Proliferation: Widespread

Description:
Troop Capacity: 2 in front, 5 in rear
Weight (mt):
Gross Vehicle Weight: 2.4
Curb: 1.6
Length Overall (m): 4
Height Overall (m): 2
Width Overall (m): 1.8
Payload on/off Highway (kg): 600
Number of Axles: 2
Ground Clearance (mm): 300
Turning Radius (m): 6.5
Wheels:
Size (in): 8.40x15
Central Tire Pressure Regulation System: No
Run Flat: No

AUTOMOTIVE

Engine: Inline 4, 70 hp, gasoline
Cooling: Water
Cruising Range (road) (km): 730
Speed (km/h): 100
Fuel Capacity (liters):
Left Tank: 39
Right Tank: 39
Towing Capability (kg):
Off Highway: 850
On Highway: 850
Gradient (loaded) (°): 60
Fording Depths (m): .58
Trench Crossing Width (mm): INA

CARGO SPACE

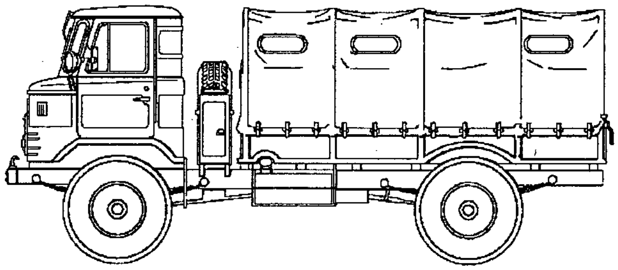
Height (mm): 400
Width (mm): 1,400
Length (mm): 1,000
Cargo Bed Area (m^2): 1.6

VARIANTS

UAZ-469B: Export; lower off-road performance
Ambulance: 1 stretcher patient and 1 attendant

NOTES
The UAZ-469 replaces the earlier UAZ-69.

Russian 2 mt 4 x 4 Cargo Truck GAZ-66

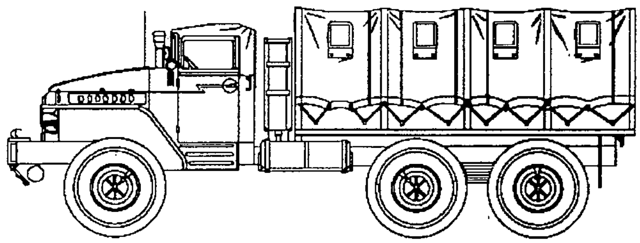


SYSTEM	AUTOMOTIVE
Alternative Designations: INA **Date of Introduction:** 1964 **Proliferation:** Widespread **Description:** Troop Capacity: 3 in cab, 18 in rear Weight (mt): Gross Vehicle Weight: 5.8 Curb: 3.6 Length Overall (m): 5.65 Height Overall (m): 2.44 Width Overall (m): 2.32 Payload on/off Highway (kg): 2,000 Number of Axles: 2 Ground Clearance (mm): 315 Turning Radius (m): 10 Wheels: Size (in): 12x18 Central Tire Pressure Regulation System: Yes	**Engine**: V8, 115 hp, gasoline **Cooling:** Water **Cruising Range (road) (km):** 875 **Speed (km/h):** 95 **Fuel Capacity (liters):** Left Tank: 105 Right Tank: 105 **Towing Capability (kg):** Off Highway: 2,000 On Highway: 2,000 **Gradient (loaded) (°):** 39 **Fording Depths (m):** .8 **CARGO SPACE** **Height (mm):** 890 **Width (mm):** 2,050 **Length (mm):** 3,330 **Cargo Bed Area (m^2):** 6.8 **VARIANTS** **GAZ-66B:** Canvas-top cab for air transport or airdrop **GAZ-66A:** Steelcab Numerous other variants for various duties.

NOTES

Besides functioning as a general cargo carries, the GAZ-66 is used as a prime mover for 120-mm mortar. The DDA-66 variant is an NBC decontamination truck.

Russian 4.5 mt 6 x 6 Cargo Truck Ural-375D



SYSTEM
Alternative Designations: INA
Date of Introduction: 1965
Proliferation: Widespread

Description:
Troop Capacity: 3 in cab, 24 in rear
Weight (mt):
  Gross Vehicle Weight: 13.2
  Curb: 8.4
Length Overall (m): 7.36
Height Overall (m): 2.68
Width Overall (m): 2.67
Number of Axles: 3
Ground Clearance (mm): 410
Turning Radius (m): 10.8
Side Slope (°): 32
Vertical Step (mm): 800
Gradient (loaded) (°): 65
Fording Depths (m): 1.49
Trench Crossing Width (mm): 875
Wheels:
  Size (in): 14x20
  Central Tire Pressure Regulation System: Yes
  Run Flat: INA

AUTOMOTIVE

Engine: V8, 180 hp, gasoline
Cooling: Liquid
Cruising Range (road) (km): 650
Speed (km/h): 75
Fuel Capacity (liters):
Main Tank: 300
Aux Tank: 60
Towing Capability (kg):
Off Highway: 5,000
On Highway: 10,000

CARGO SPACE

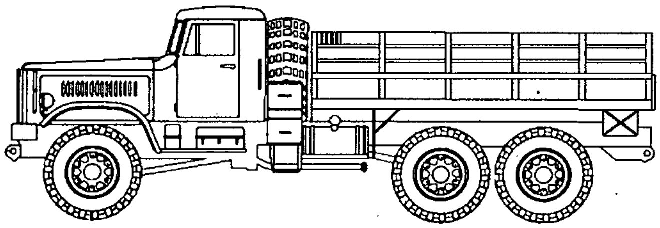
Height (mm): 872
Width (m): 2.43
Length (m): 3.9
Cargo Bed Area (m^2): 9.5

VARIANTS

URAL-375: Observation hatch and unimproved powertrain
URAL-375E: Decontamination vehicle
URAL-375N: 2,000 kg additional payload
URAL-375S: Truck-tractor
URAL-375T: Equipped with winch

NOTES
Besides functioning as a general cargo carrier, the Ural-375D is used as a prime mover for light and medium artillery. The Ural-375 chassis also serves as a base for the BM-21 MRL, POL tankers, vans, and cranes. The Ural-4320 began to replace the Ural-375D around 1978.

Russian 7.5 mt 6 x 6 Cargo Truck KrAZ-255B



SYSTEM

Alternative Designations: INA
Date of Introduction: 1967
Proliferation: Widespread

Description:
Troop Capacity: 3 in cab, 16 in rear
Weight (mt):
Gross Vehicle Weight: 19.7
Curb: 12
Length Overall (m): 8.64
Height Overall (m): 2.94
Width Overall (m): 2.75
Payload (kg): 7,500
Number of Axles: 3
Ground Clearance (mm): 360
Turning Radius (m): 14
Wheels:
Size (in): 20x21
Central Tire Pressure Regulation System: Yes
Run Flat: INA

AUTOMOTIVE

Engine: V8, 265 hp, diesel
Cooling: Water
Cruising Range (road) (km): 850
Speed (km/h): 70
Fuel Capacity (liters):
Right Tank: 165
Left Tank: 165
Towing Capability (kg):
Off Highway: 10,000
On Highway: 50,000
Gradient (loaded) (°): 30
Fording Depths (m): 1

CARGO SPACE

Length (m): 4.56
Width (m): 2.5
Height (m): .92
Cargo Bed Area (m^2): INA

VARIANTS

KrAZ-258: Tractor-truck
Numerous other variants for various duties.

NOTES
Primarily designed as a cargo truck, the KrAZ-255B is also used as a prime mover for various equipment including a tank-transporter trailer and PMP pontoon bridge.

Chapter 9
Rotary-Wing Aircraft

This chapter provides the basic characteristics of selected rotary-wing aircraft readily available to the OPFOR. Both FM 100-60, *Armor- and Mechanized-Based Opposing Force: Organization Guide* and FM 100-63, *Infantry-Based Opposing Force: Organization Guide*, use generic descriptors to indicate helicopter capabilities. This enables the trainer to structure OPFOR air support requirements by capability rather that specific equipment type. ***Rotary-Wing Aircraft***, cover systems classified as light, attack, utility, and heavy aircraft systems. Some multi-role aircraft will be able to support missions across each of the categories. Therefore, they are listed in each of the above categories by their initial design, and their planned application. This chapter encompasses many aircraft which may have a dual civil/military history. It does not include however, aircraft designed and used primarily for civil aviation.

This initial sampling of systems was selected because of their wide proliferation across numerous countries or because of their already extensive use in training scenarios. Additional data sheets addressing other widely proliferated helicopter systems will be sent with further supplements to this guide.

Because of the increasingly large numbers of variants of each aircraft, only the most common variants produced in significant numbers were addressed. If older versions of helicopters have been upgraded in significant quantities to the standards of newer variants, the older versions were not addressed.

The munitions available to each aircraft are mentioned, but not all may be employed at the same time. The weapon systems inherent to the airframe are listed under armament. The most probable weapon loading options are also given, but assigned mission dictates actual weapon configuration. Therefore, any combination of the available munitions may be encountered.

Chapter 10, ***Fixed-Wing Aircraft***, will be constructed with future supplements to this guide. It will provide the basic characteristics of selected fixed-wing aircraft readily available to the OPFOR. It will initially focus on the aircraft commonly employed by the OPFOR when in close proximity to enemy ground forces. Sample aircraft included will be categorized by the missions of reconnaissance, interdiction, strike, direct air support, and transport.

Questions and comments on data listed in this chapter should be addressed to:

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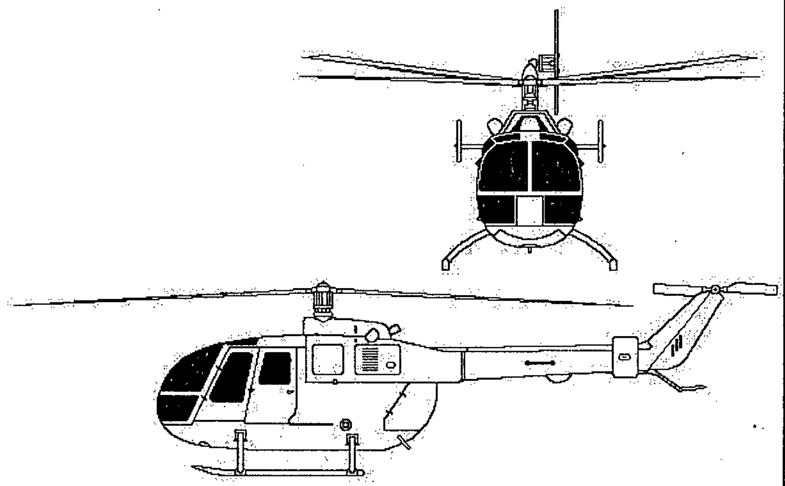
European Light Helicopter BO-105



Weapon & Ammunition Types	Combat Load
Other Loading Options	
7.62-mm or 12.7-mm MG pods	
2.75-in rocket pods (7 or 12 ea.)	2
68-mm SNEB rocket pods (12ea)	2
50-mm SNIA rockets (28 ea.)	2
TOW ATGM pods (4 ea.)	2
HOT ATGM	6
AS-12 ASM pods (2 ea.)	2
Stinger AAM pod (4 ea.)	1

SYSTEM

Alternative Designations: INA
Date of Introduction: 1972
Proliferation: At least 40 countries

Description: Variants in "( )"
Crew: 1 or 2 (pilots)
Blades:
  Main rotor: 4
  Tail rotor: 2
Engines: 2x 420-shp Allison 250-C20B turboshaft
Weight (kg):
  Maximum Gross: 2,500
  Normal Takeoff: 2,000
  Empty: 1,301, 1,913 (PAH1)
Speed (km/h):
  Maximum (level): 242
  Cruise: 205
Ceiling (m):
  Service: 3,050
  Hover (out of ground effect): 457
  Hover (in ground effect): 1,525
Vertical Climb Rate (m/s): 7.5
Fuel (liters):
  Internal: 570
  Internal Aux Tank: 200 ea. (max 2x)
Range (km):
  Normal Load: 555
  With Aux Fuel: 961
Dimensions (m):
  Length (rotors turning): 11.9
  Length (fuselage): 8.8
  Width: 2.5
Dimensions continued (m):
  Height: 3.0
  Main Rotor Diameter: 9.8
  Tail Rotor Diameter: 1.9
Cargo Compartment Dimensions (m):
  Floor Length: 1.9
  Width: 1.4
  Height: 1.3
Standard Payload (kg):
  Internal load: 690
  External on sling only: 1,200
  Transports 3 troops or 2 litters, or cargo.

Survivability/Countermeasures:
Main and tail rotors electrically deiced.
Infrared signature suppressors can be mounted on engine exhausts.
Rotor brake.

ARMAMENT

Most Probable Armament:
BO-105P/PAH1: Outriggers carry 6x HOT antitank missiles, or rocket pods.

CASA BO-105/ATH: The Spanish produced variant rigidly mounts 1x Rh 202 20-mm cannon under the fuselage.

AVIONICS/SENSOR/OPTICS

The BO-105P has a roof-mounted direct-view, daylight-only sight to allow firing of HOT ATGMs. Options exist to fit a thermal imaging system for night operations, and a laser designator.

Night/Weather Capabilities:
Available avionics include weather radar, Doppler and GPS navigation, and an autopilot. It is capable of operation in day, night, and instrument meteorological conditions.

VARIANTS

The BO 105 was developed initially by Messerschmitt-Bolkow-Blohm in Germany. Others are built in Chile, the Philippines, Indonesia (**NBO-105**), and Spain (**CASA BO-105/ATH**).

BO-105CB: The standard production variant.

BO-105CBS: VIP version with a slightly longer fuselage to accommodate 6 passengers, some used in a SAR role.

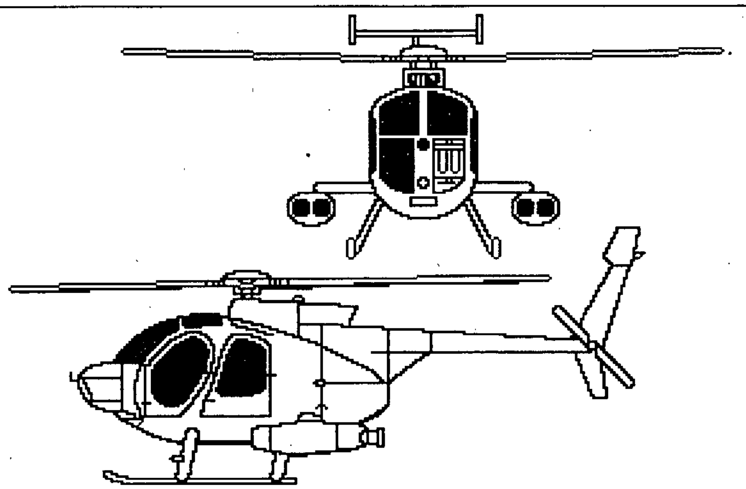
BO-105LS: Upgraded to 2x 550-shp Allison 250-C28 turboshaft engines for extended capabilities in high altitudes and temperatures. Produced only in Canada.

BO-105M/VBH: Standard reconnaissance version.

BO-105P/PAH1: Standard antitank version.

NOTES
Available munitions are shown above; not all will be employed at the same time, mission dictates weapons configuration. External stores are mounted on weapons "outriggers" or racks on each side of the fuselage. Each rack has one hardpoint. This helicopter is produced by the Eurocopter Company. It was formed as a joint venture between Aerospatiale of France, and Daimler-Benz Aerospace of Germany. Other missions include: direct air support, antitank, reconnaissance, search and rescue, and transport. Clamshell doors at rear of cabin area open to access cargo area. Cargo floor has tiedown rings throughout.

United States Light Helicopter MD-500/Defender



Weapon & Ammunition Types Other Loading Options	Combat Load
M134 7.62-mm 6x barrel, Gatling type twin MG pods	2000
M260 2.75-in Hydra 70 rocket pods (7 or 12 each)	2
.50 cal MG pods	2
M75 40-mm grenade launchers	2
MK19 40-mm grenade launcher	2
TOW missile pods (2 each)	2
Hellfire ATGM	
Stinger AAM	

SYSTEM

Alternative Designations: Hughes model 369, Cayuse, Loach
Date of Introduction: 1977 (MD-500 MD)
Proliferation: At least 22 countries

Description: Variants in "( )"
Crew: 1 or 2 (pilots)
Blades:
Main rotor: 4 or 5 (see VARIANTS)
Tail rotor: 2 or 4 (see VARIANTS)
Engines: (see VARIANTS)
Weight (kg):
Maximum Gross: 1,361 (500), 1,610 (530)
Normal Takeoff: 1,090
Empty: 896
Speed (km/h):
Maximum (level): 241 (500), 282 (530)
Cruise: 221 (500), 250 (530)
Ceiling (m):
Service: 4,635 (500), 4,875 (530)
Hover (out of ground effect): 1,830 (500), 3,660 (530)
Hover (in ground effect): 2,590 (500), 4,360 (530)
Vertical Climb Rate (m/s): 8.4 (500), 10.5 (530)
Fuel (liters):
Internal: 240
Internal Aux Tank: 80
Range (km):
Normal Load (est.): 485 (500), 430 (530)
Dimensions (m):
Length (rotors turning): 9.4 (500), 9.8 (530)
Length (fuselage): 7.6 (500), 7.3 (530)
Width: 1.9
Height: 2.6 (500), 3.4 (530 over mast-mounted sight)

Dimensions continued (m):
Main Rotor Diameter: 8.0 (500), 8.3 (530)
Tail Rotor Diameter: 1.4
Cargo Compartment Dimensions (m):
Floor Length: 2.4
Width: 1.3
Height: 1.5
Standard Payload (kg):
Internal load: INA
External load: 550
Transports 2 or 3 troops or cargo internally, or 6 on external platforms in lieu of weapons.

Survivability/Countermeasures:
Some models have radar warning receivers.
Chaff and flare systems available.
Infrared signature suppressors can be mounted on engine exhausts.

ARMAMENT

Most Probable Armament: (MD-500D pictured)
MD-500MD/Scout Defender: Fitted with guns, rockets, grenade launchers, or a combination on 2x fuselage hardpoints.

MD-500MD/TOW Defender: Twin TOW missile pods on 2x fuselage hardpoints; mounts missile sight in lower-left front windshield.

AVIONICS/SENSOR/OPTICS

The MD-500 allows for the mounting of a stabilized, direct-view optical sight in the windshield. Options exist to fit a mast-mounted, multiple field of view optical sight, a target tracker, a laser rangefinder, thermal imager, a 16x FLIR for night navigation and targeting, and autopilot.

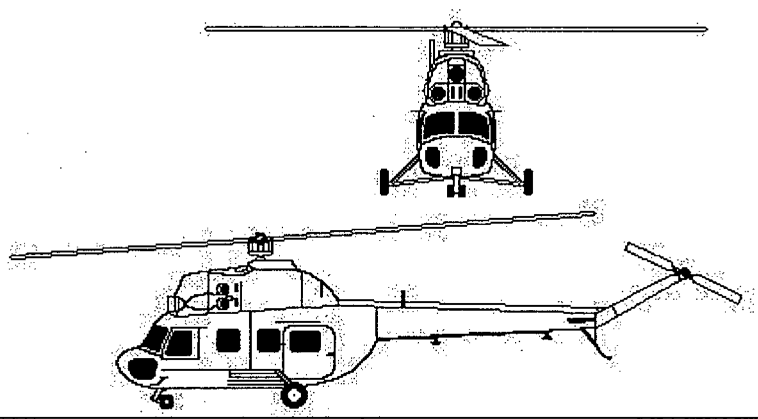
Night/Weather Capabilities:
Optional avionics include GPS, ILS and full instrument weather conditions packages.
The more advanced variants are fully capable of performing all missions under any conditions.

VARIANTS

OH-6A/Cayuse: Developed initially by the Hughes Aircraft company (later McDonnell Douglas Helicopter Company) in the mid-1960s for the US Army. Fitted with 1x 253-shp Allison T63-A-5A turboshaft, 4 bladed main rotor, and an offset "V" tail.
Hughes 500M: Military export version of OH-6 in mid-1970s with upgraded 278-shp Allison 250-C18 turboshaft engine, "V" tail.
MD-500MD/Scout and TOW Defender: Improved military version of the model 500 with 5 main rotor blades, 375-shp Allison 250-C20B turboshaft engine, and T-tail.
MD-500E/MD-500MG/Defender II: Had a more elongated nose for streamlining, and an optional 4x blade tail rotor for reduced acoustic signatures. Possible mast-mounted sight.
OH-6A/MD-530F Super Cayuse/Lifter: Upgraded engine to a 425-shp Allison 250-C30 turboshaft, and avionics in 1988 for the US Army.
MD-530MG/Defender: Has a mast-mounted sight, and incorporated upgrades of all previous variants.
AH/MH-6J: US Army Special Operations variant derived from the MD-530MG.

NOTES
Available munitions are shown above; not all will be employed at the same time, mission dictates weapons configuration. External stores are mounted on weapons racks on each side of the fuselage. Each rack has one hardpoint. Other missions include: direct air support, antitank, reconnaissance, observation, and light utility.

Russian Light Helicopter Mi-2/HOPLITE



Weapon & Ammunition Types	Combat Load
1x 23-mm automatic cannon	
1x 7.62-mm or 12.7-mm MG	
Other Loading Options:	
AT-3c/SAGGER ATGM	4
57-mm Rocket pods (16 each)	2
Twin or single fixed 7.62-mm or 12.7-mm MG	
External fuel tanks (liters)	238
SA-7b/GRAIL missile	4

SYSTEM

Alternative Designations: INA
Date of Introduction: 1965
Proliferation: Widespread

Description:
Crew: 1 (pilot)
Blades:
  Main rotor: 3
  Tail rotor: 2
Engines: 2x 400-shp PZL GTD-350 (series III and IV) turboshaft
Weight (kg):
  Maximum Gross: 3,700
  Normal Takeoff: 3,550
  Empty: 2,372
Speed (km/h):
  Maximum (level): 220
  Cruise: 194
Ceiling (m):
  Service: 4,000
  Hover (out of ground effect): 1,000
  Hover (in ground effect): 2,000
Vertical Climb Rate (m/s): 4.5
Fuel (liters):
  Internal: 600
  Internal Aux Tank: N/A
  External Fuel Tank: 238 ea.
Range (km):
  Maximum Load: 580
  Normal Load: 340
  With Aux Fuel: 790

Dimensions (m):
  Length (rotors turning): 17.4
  Length (fuselage): 11.9
  Width: 3.2
  Height: 3.7
  Main Rotor Diameter: 14.6
  Tail Rotor Diameter: 2.7
Standard Payload:
  Transports 6-8 troops or 700 kg internal cargo or 800 kg external load on 4x external hardpoints.

Survivability/Countermeasures:
Main and tail rotor blades electrically deiced.

ARMAMENT
23-mm Automatic Cannon, NS-23KM:
Range: (practical) 2,500 m
Elevation/Traverse: None (rigidly-mounted)
Ammo type: HEFI, HEI, APT, APE, CC
Rate of Fire (rpm): (practical) 550

7.62-mm or Pintle-mounted Machinegun:
(may be mounted in left-side cabin door)
Range: (practical) 1,000 m
Ammo type: HEFI, HEI, APT, APE, CC
Rate of Fire (rpm): (practical) 250

OR

12.7-mm or Pintle-mounted Machinegun:
(may be mounted in left-side cabin door)
Range: (practical) 1,500 m
Ammo type: API, API-T, IT, HEI
Rate of Fire (rpm): (practical) 100

AVIONICS/SENSOR/OPTICS

The cannon is pilot sighted, and fire is adjusted by controlling the attitude of the aircraft.

Night/Weather Capabilities:
The Mi-2 is primarily a daylight only aircraft.

VARIANTS

Mi-2R: Ambulance version that carries 4x litter patients.

Mi-2T: Transport version that carries 8 personnel.

Mi-2URN: Armed reconnaissance variant, employs 57-mm unguided rockets, and mounts a gunsight in the cockpit for aiming all weapons.

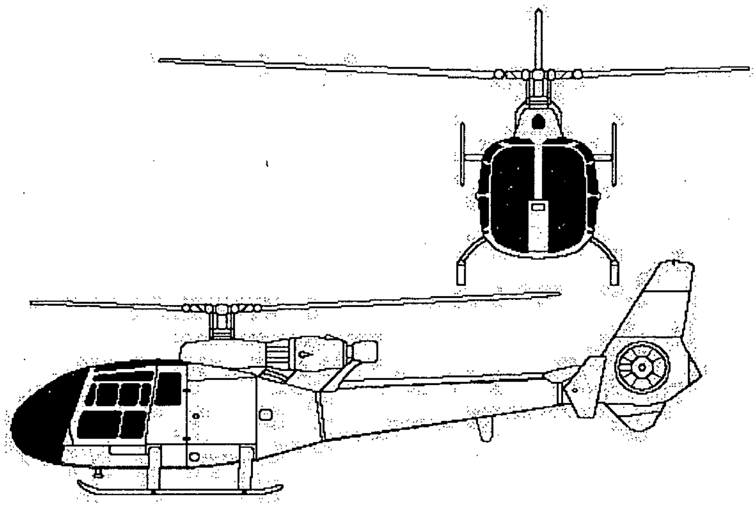
Mi-2URP: The antitank variant. Carries 4x AT-3 Sagger wire-guided missiles on external weapons racks, and 4x additional missiles in the cargo compartment.

Mi-2US: The gunship variant, employs an airframe modification that mounted a 23-mm NS-23KM cannon to the portside fuselage. Also employs 2x 7.62-mm gun pods on external racks, and 2x 7.62-mm pintle-mounted machineguns in the cabin.

PZL Swidnik: A Polish-produced variant under license from Russia. Same performance, characteristics, and missions.

NOTES
Available munitions are shown above; not all will be employed at the same time, mission dictates weapons configuration. External stores are mounted on weapons racks on each side of the fuselage. Each rack has two hardpoints for a total of four stations. Additional missions include; direct air support, antitank, armed reconnaissance, transport, medevac, airborne command post, smoke generating, minelaying, and training. The cabin door is hinged rather than sliding, which may limit operations. There is no armor protection for the cockpit or cabin. Ammo storage is in the aircraft cabin, so combat load varies by mission. Some Mi-2USs currently employ fuselage-mounted weapon racks rather than the 23-mm fuselage-mounted cannon which is removed. Some variants however, still employ the cannon.

French Light Helicopter SA-341/GAZELLE



Weapon & Armament Types	Combat Load
7.62-mm MG or	
20-mm GIAT M.621 cannon or	100
2x 7.62-mm AA-52 FN MG pods	1,000
Other Loading Options	
2.75-in rocket pods (7 ea.)	2
68-mm SNEB rocket pods (12 ea)	2
57-mm rockets (18 ea.)	2
HOT ATGM	4-6
AT-3 SAGGER ATGM	4
AS-11 ASM, or AS-12 ASM	4 or 2
SA-7 GRAIL AAM	2
MISTRAL AAM	2

SYSTEM

Alternative Designations: SA-342
Date of Introduction: 1973
Proliferation: At least 23 countries

Description: Variants in "( )"
Crew: 1 or 2 (pilots)
Blades:
Main rotor: 3
Tail rotor: 13 (fenestron enclosed in tail)
Engines: 1x 590-shp Turbomeca Astazou IIIB turboshaft
Weight (kg):
Maximum Gross: 1,800 (SA 341), 1,900 (SA 342K), 2,000 (SA 342L/M)
Normal Takeoff: 1,800
Empty: 998
Speed (km/h):
Maximum (level): 310
Cruise: 270
Ceiling (m):
Service: 4,100 (SA 341), 5,000 (SA 342)
Hover (out of ground effect): 2,000 (SA 341), 2,370 (SA 342)
Hover (in ground effect): 2,850 (SA 341), 3,040 (SA 342)
Vertical Climb Rate (m/s): 12.2
Fuel (liters):
Internal: 445
Internal Aux Tank: 90
Additional Internal Aux Tank: 200
Range (km):
Normal Load: 670 (SA 341), 735 (SA 342)

Dimensions (m):
Length (rotors turning): 11.9
Length (fuselage): 9.5
Width: 2.0
Height: 3.1
Main Rotor Diameter: 10.5
Tail Rotor Diameter: 0.7
Cargo Compartment Dimensions (m):
Floor Length: 2.2
Width: 1.3
Height: 1.2
Standard Payload (kg):
Internal load: 750
External on sling only: 700
Transports 3 troops or 1 litter, or cargo.

Survivability/Countermeasures:
IR signature suppressor on engine exhaust.

ARMAMENT

Most Probable Armament:
SA 341F: A GIAT M.621 20-mm cannon is installed on starboard side of some aircraft. Rate of fire is selectable at 300 or 740 rpm.
SA 341H: Can carry 4x AT-3 ATGMs, and 2x SA-7, or 128-mm or 57-mm rockets, and 7.62-mm machinegun in cabin.
SA 342K: Armed antitank version with 4-6x HOT ATGMs.
SA 342L: Either rocket pods or machineguns.
SA 342M: Armed with 4-6x HOT antitank missiles, and possibly fitted with Mistral air to air missiles.

AVIONICS/SENSOR/OPTICS

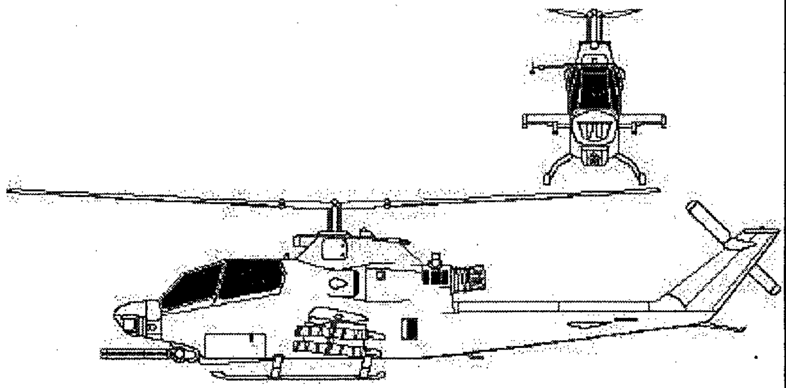
The SA 342M has a roof-mounted stabilized direct view/infrared/laser sight to allow night firing of HOT ATGMs.

Night/Weather Capabilities:
The aircraft is NVG compatible; and by its instruments, avionics, autopilot, and nav computer, is capable of flight in day, night, and instrument meteorological conditions.

VARIANTS
AS 341 Gazelle: Developed by Aerospatiale in France. Others were built in the UK by Westland, and in Yugoslavia.
SA 341B/C/D/E: Production versions for the British military. Used in training and communications roles.
SA 341F: Production version for the French Army. Upgraded engine to Astazou IIIC.
SA 341H: Export variant.
SA 342K: Armed SA 341F with upgraded 870-shp Astazou XIVH engine, mostly exported to the Middle East.
SA 342L: Export light attack variant with Astazou XIVM engine.
SA 342M: Improved ground attack variant for the French Army. Similar to SA 342L, but with improved instrument panel, engine exhaust baffles to reduce IR signature, navigational systems, Doppler radar, and other night flying equipment.

NOTES
Available munitions are shown above; not all will be employed at the same time, mission dictates weapons configuration. External stores are mounted on weapons "outriggers" or racks on each side of the fuselage. Each rack has one hardpoint. Other missions include: attack, antitank, antihelicopter, reconnaissance, utility, transport, and training. The bench seat in the cabin area can be folded down to leave a completely open cargo area. Cargo floor has tiedown rings throughout.

United States Attack Helicopter AH-1F/COBRA



Weapon & Ammunition Types	Combat Load
20-mm 3x barrel Gatling gun	**750**
Other Loading Options	
TOW missile pods (4 each)	**0-2**
2.75-in Hydra 70 rocket pods (19 each)	**2-4**
7.62-mm 6x barrel rotary MG pods	**0-2**

SYSTEM

Alternative Designations: Hueycobra, Bell 209
Date of Introduction: 1986 (AH-1S)
Proliferation: At least 11 countries

Description:
Crew: 2 (pilots in tandem seats)
Blades:
  Main rotor: 2
  Tail rotor: 2
Engines: 1x 1,800-shp AlliedSignal Engines T-53-L-703 turboshaft
Weight (kg):
  Maximum Gross: 4,535
  Normal Takeoff: 4,524
  Empty: 2,993
Speed (km/h):
  Maximum (level): 315
  Cruise: 227
Max "G" Force: INA
Ceiling (m):
  Service: 3,720
  Hover (out of ground effect): INA
  Hover (in ground effect): 3,720
Vertical Climb Rate (m/s): 8.5
Internal Fuel (liters): 991
Range (km):
  Normal Load: 610
  With Aux Fuel: N/A
Dimensions (m):
  Length (rotors turning): 16.3
  Length (fuselage): 13.6
  Width (including wing): 3.2
  Height: 4.1
  Main Rotor Diameter: 13.4
  Tail Rotor Diameter: 2.6
Cargo Compartment Dimensions: negligible
Standard Payload (kg): 1,544

Survivability/Countermeasures:
Infrared signature suppressors mounted on engine exhaust.
Radar warning receivers, IFF, Infrared jammer, chaff and flares.
Armored cockpit.

ARMAMENT
The chin-mounted turret accepts Gatling-type guns ranging from 7.62-mm to 30-mm.
Some aircraft have been modified to accept Stinger missiles (air-to-air Stinger or ATAS).

20-mm 3x barrel Gatling gun, M197:
Range: (practical) 1,500 m
Elevation: 21° up to 50° down
Traverse: 220°
Ammo Type: AP, HE
Rate of Fire: burst 16±4, continuous 730±50

Most Probable Armament:
AH-1G: Either 2x 7.62-mm miniguns with 4,000 rounds or 2x 40-mm grenade launchers with 300 rounds (one each is possible) in chin turret. Also on underwing hardpoints, 2.75-in. FFAR, minigun pods, or 20-mm automatic cannons.

AH-1S: M197, 3x barrel 20-mm Gatling gun in chin turret. Also on underwing hardpoints, 8x BMG71 TOW antitank missiles, and 2x 2.75-in FFAR rocket pods.

AVIONICS/SENSOR/OPTICS
The TOW missile targeting system uses a telescopic sight unit (traverse 110°, elevation –60°/+30°), a laser augmented tracking capability, thermal sights and a FLIR to allow for acquisition, launch, and tracking of all types of TOW missiles in all weather conditions.
The Cobra also uses a digital ballistic computer, a HUD, Doppler nav, and a low speed air data sensor on the starboard side for firing, and has in-flight boresighting.
Available Israeli-made upgrades include an integrated FLIR with laser rangefinder, GPS, automatic boresighting, and the ability to fire both TOW II and Hellfire missiles.

Night/Weather Capabilities:
The AH-1 is fully capable of performing its attack mission in all weather conditions.

VARIANTS
Most older Cobra variants still in operation have been upgraded to the AH-1F standard.
Also produced in Romania and Japan under license from Bell Textron in the U.S.

AH-1G: Initial production model in 1966

AH-1S: Upgraded 1960s produced aircraft in late 1980s to the standard TOW carrying version.

AH-1P: A set of AH-1S aircraft fitted with composite rotors, flat plate glass cockpits, and NVG capabilities.

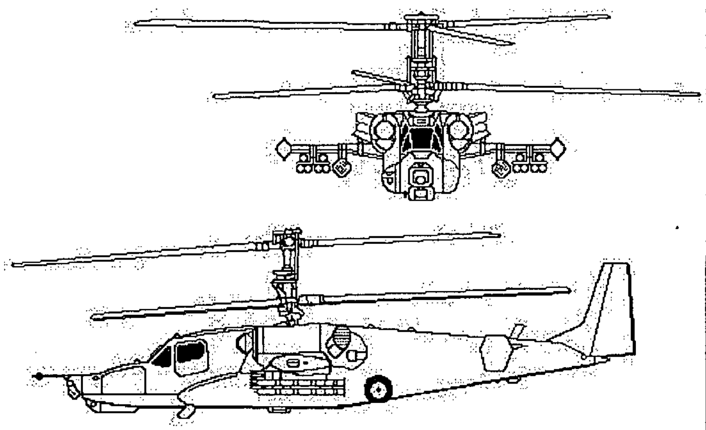
AH-1E: A set of AH-1S aircraft upgraded with the Enhanced Cobra Armament System incorporating the universal turret, 20-mm gun, automatic compensation for off-axis gun firing, and weapon management system.

AH-1F: Current standard Cobra. Also referred to as the "Modernized Cobra". Incorporated all past upgrades.

AH-1J/-1T/-1W: See separate AH-1W entry.

NOTES
Available munitions are shown above; not all may be employed at one time. Mission dictates weapon configuration. External stores are mounted on underwing external stores points. Each wing has two hardpoints for a total of four stations. A representative mix when targeting armor formations would be eight TOW missiles, two 2.75-in rocket pods, and 750x 20-mm rounds. The gun must be centered before firing underwing stores. Additional missions include direct air support, antitank, armed escort, and air to air combat. Armored cockpit can withstand small arms fire, and composite blades and tailboom are able withstand damage from 23-mm cannon hits.small arms fire, and composite blades and tailboom able to withstand damage from 23-mm cannon hits.

Russian Attack Helicopter Ka-50/HOKUM



Weapon & Ammunition Types	Combat Load
1x 2A42 30-mm cannon	
HE-Frag	250
AP	250
Total	**500**
Other Loading Options	
AT-16 VIKhR ATGM (6 each)	**2**
80-mm rockets (20 each)	**2**
Twin 23-mm gun pods	**940**
500-kg bombs	**4**
AA-11/ARCHER AAM	**2**
External fuel tanks (liters)	**500**

SYSTEM

Alternative Designations: Black Shark, Werewolf
Date of Introduction: N/A
Proliferation: Preproduction. An initial fielding plan is for 2 per year for 14 years.

Description:
Crew: 1 (pilots, 2 in Ka-52)
Blades:
Main rotor: 6 (2 heads, 3 blades each)
Tail rotor: None
Engines: 2x 2,200-shp Klimov TV3-117VK *turboshaft*
Weight (kg):
Maximum Gross: 10,800
Normal Takeoff: 9,800
Empty: 7,692
Speed (km/h):
Maximum (level): 340 (est.)
Cruise: 270
Sideward: 100+, Rearward: 100+
Turn Rate: unlimited
Max "G" Force: +3 to +3.5 g
Ceiling (m):
Service: 5,500
Hover (out of ground effect): 4,000
Hover (in ground effect): 5,500
Vertical Climb Rate (m/s): 10
Fuel (liters):
Internal: INA
External Fuel Tank: 500 ea. (max 4x)
Range (km):
Maximum Load: INA
Normal Load: 460
With Aux Fuel: INA
Dimensions (m):
Length (rotors turning): 16
Length (fuselage): 15.0
Width (including wing): 7.34
Height (gear extended): 4.93
Height (gear retracted): 4
Main Rotor Diameter: 14.5
Cargo Compartment Dimensions: Negligible
Standard Payload:
External weapons load: 2,500 kg on 4 underwing stores points.

Survivability/Countermeasures:
Main rotors and engines electrically deiced.
Infrared signature suppressors can be mounted on engine exhausts.
Radar warning receivers, IFF, chaff and flares.
Armored cockpit and self-sealing fuel tanks.
Pilot ejection system.
(see NOTES)

ARMAMENT
30-mm Automatic Cannon, 2A42:
Range: effective 3,000 m
Elevation: -45° to +10°
Traverse: ±15°
Ammo type and rate of fire is selectable by pilot (HE or AP, 350 or 600)

Most Probable Armament: (shown above)
HOKUM A/N: Fuselage-mounted 30-mm cannon on right side, 80-mm rockets, AT-16 VIKhR ATGMs.

HOKUM B: Same as above.

ATGM, AT-16/VIKhR:
Guidance: Laser Beam Rider SACLOS
Range: 10,000 m
Warhead: HEAT
Penetration: 900 mm
Effective against ground & air targets at converging speeds to 800 km/h.
ATGM racks can depress to 12°.

AVIONICS/SENSOR/OPTICS

The HOKUM uses a low-light level TV or thermal sighting, a laser range-finder (10 km), FLIR, air data sensor, and digital datalink which interface with a fire control computer, an autopilot, a helmet sighting system and HUD for target location, acquisition, designation, and firing.

Night/Weather Capabilities:
This aircraft's avionics package ensuring a full day/night, all weather capability. If it is to be employed at night in an attack role, it *must be fitted with a night targeting pod.* This pod includes a FLIR, a millimeter wave radar, and an electro-optical sight takes up one of the underwing pylons.
The Ka-50N, and Ka-52 are capable of performing attack missions in day/night, and all-weather conditions.
The French companies Thomson-CSF, and Sextant Avionique offer nav/attack systems, which can be fitted to export variants.

VARIANTS

Ka-50A/HOKUM A: Standard direct air support variant.

Ka-50N/HOKUM N: Night attack variant fitted with a nose-mounted FLIR. The cockpit is fitted with an additional TV display, and is NVG compatible.

Ka-52/HOKUM B: The "Alligator" is a side-by-side, two-seat cockpit variant of the Ka-50. The gross weight of the aircraft is greater, so the performance is marginally degraded. But airframe characteristics, dimensions, and armaments are relatively similar. It includes a mast-mounted millimeter wave radar covering the front quadrant only. It is used as an attack aircraft, and as a trainer for the Ka-50.

Russian Attack Helicopter Ka-50/HOKUM continued

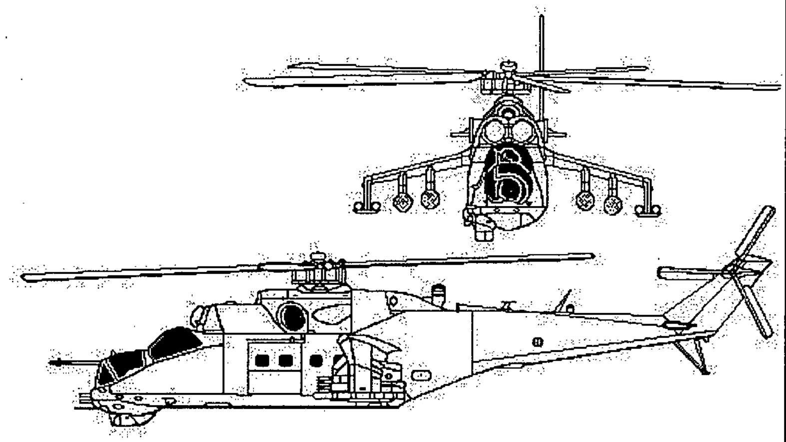
NOTES

This aircraft is not fielded. Only a handful of prototypes exist, and it has not yet been approved for full-scale production.

The fully armored pilot's cabin can withstand 23-mm gunfire, and the cockpit glass 12.7-mm MG gunfire. The Zvezda K-37-800 pilot ejection system functions at any altitude. Available munitions are shown above; not all may be employed at one time. Mission dictates weapons configuration. External stores are mounted on underwing external hardpoints. Each wing has two hardpoints for a total of four stations. A typical mix for targeting armor formations is 12x AT-16 ATGMs, 500x 30-mm cannon rounds, and 2x 20-round pods of 80-mm folding fin unguided rockets. It was designed for remote operations, and not to need ground maintenance facilities for 2 weeks. The 30-mm cannon is the same as on the BMP-2. The firing computer will turn the aircraft to keep the gun on target. A coaxial counter-rotating rotor system negates the need for a tail rotor and its drive system. Because of this, this aircraft is unaffected by wind strength and direction, has an unlimited hovering turn rate, and gives a smaller profile and acoustic signature, while allowing a 10-15% greater power margin. The airframe is 35% composite materials with a structural central 1m^2 keel beam of kevlar/nomex that protects critical systems and ammunition. The HOKUM is fully aerobatic. It can perform loops, roll, and "the funnel", where the aircraft will maintain a concentrated point of fire while flying circles of varying altitude, elevation, and airspeed around the target.

Russian Attack Helicopter Mi-24/HIND

	Weapon & Ammunition Types	Combat Load



1x twin 30-mm gun or	750
12.7-mm 4 barrel turret gun	1,470
Other Loading Options	
AT-2C or AT-6C ATGMs	2-12
80-mm S-8 rocket pods (20 ea.)	2-4
57-mm S-5 rocket pods (32 ea.)	2-4
GSh-23L twin 23-mm MG pods	940
250-kg bombs	4
500-kg bombs	2
External fuel tanks (liters)	500

SYSTEM

Alternative Designations: INA
Date of Introduction: 1976 (HIND D)
Proliferation: At least 34 countries

Description:
Crew: 2 (pilots in tandem cockpits)
Blades:
  Main rotor: 5
  Tail rotor: 3
Engines: 2x 2,200-shp Klimov TV3-117VMA turboshaft
Weight (kg):
  Maximum Gross: 11,500
  Normal Takeoff: 11,100
  Empty: 8,500
Speed (km/h):
  Maximum (level): 335
  Cruise: 295
Max "G" Force: 1.75 g
Ceiling (m):
  Service: 4,500
  Hover (out of ground effect): 1,500
  Hover (in ground effect): 2,200
Vertical Climb Rate (m/s): 15
Fuel (liters):
  Internal: 1,840
  Internal Aux Tank (in cabin): 1,227
  External Fuel Tank: 500 ea.
Range (km):
  Normal Load: 450
  With Aux Fuel: 950
Dimensions (m):
  Length (rotors turning): 21.6
  Length (fuselage): 17.5
  Width (including wing): 6.5
  Height (gear extended): 6.5
  Main Rotor Diameter: 17.3
  Tail Rotor Diameter: 3.9
Cargo Compartment Dimensions (m):
  Floor Length: 2.5
  Width: 1.5
  Height: 1.2

Standard Payload:
  Internal load: 8 combat troops or 4 litters
  External weapons load: 1,500 kg
  External load (no weapons): 2,500 kg

Survivability/Countermeasures:
Main and tail rotors electrically deiced.
Infrared signature suppressors can be mounted on engine exhausts.
Radar warning receivers, IFF, Infrared jammer, rotor brake, chaff and flares.
Armored cockpit.

ARMAMENT
Loaded combat troops can fire personal weapons through cabin windows.

12.7-mm 4x Barrel Machinegun, YaKB-12.7:
Range (m): (practical) 1,500
Elevation/Traverse: 20° up to 60° down/ 120°
Ammo Type: HEFI, APT, Duplex, DuplexT
Rate of Fire (rpm): up to 4,500 (pilot selectable)

OR

30-mm Twin Barrel Cannon, GSh-30K:
Range (m): (practical) 4,000
Elevation/Traverse: None (rigidly mounted)
Ammo Type: HEFI, HEI, APT, APE, CC
Rate of Fire (rpm): 300, or 2,000 to 2,600

Most Probable Armament: (HIND F pictured)
HIND D: Turret-mounted 4-barrel 12.7-mm Gatling type machinegun, 57-mm rockets, AT-2C/SWATTER ATGMs.

HIND E: Turret-mounted 4-barrel 12.7-mm Gatling type machinegun or twin barrel 23-mm turret gun, 57-mm rockets, AT-6C/ SPIRAL ATGMs.

HIND F: Fixed 30-mm twin gun on the right fuselage side, 57-mm rockets, AT-6C/ SPIRAL ATGMs.

AVIONICS/SENSOR/OPTICS
The ATGM targeting system uses a low-level light TV, a laser designator, FLIR, air data sensor, and a missile guidance transmitter.

Night/Weather Capabilities:
HIND D versions are primarily daytime aircraft only. Some HIND E and Mi-35 series export versions have upgraded night and weather capabilities, better avionics, weather radar, autopilot, HUD, GPS, NVG compatibility, more armor, and an increased weapons load provided by the French company Sextant Avionique.

VARIANTS
Nearly all of the older HIND A, B and C variants have been upgraded or modified to the HIND D or E standard.

Mi-24D/HIND D: Direct air support.

Mi-24V/HIND E: Direct air support. Most proliferated version.

Mi-24P/HIND F: Direct air support. The fixed twin gun cut the turret profile, and empty weight to 8,200 kg, while boosting maximum gross weight to 12,000 kg.

Mi-24R/HIND G-1: NBC sampling. It has mechanisms to obtain soil and air samples, filter air, and place marker flares.

Mi-24K/HIND G-2: Photo-recon, and artillery spotting. Has a camera in cabin, gun, rocket pods, but no targeting system.

Mi-25: Export version of the HIND D.

Mi-35: Export version of the HIND E. The **Mi-35M** has a twin barrel 23-mm gun.

Mi-35P: Export version of the HIND F.

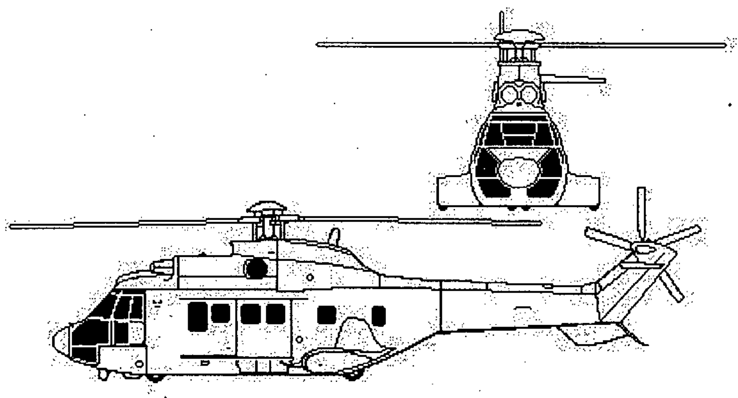
Russian Attack Helicopter Mi-24/HIND continued

NOTES
Available munitions are shown above; not all may be employed at one time. Mission dictates weapon configuration. External stores are mounted on underwing external stores points. Each wing has three hardpoints for a total of six stations. A representative mix when targeting armor formations would be eight AT-6 ATGMs, 750x 30-mm rounds, and two 57-mm rocket pods. Additional missions include direct air support, antitank, armed

escort, and air to air combat. The aircraft can store an additional ammunition basic load in the cargo compartment in lieu of carrying troops. Armored cockpits and titanium rotor head able to withstand 20-mm cannon hits. Every aircraft has an overpressurization system for operation in a NBC environment.

The HIND's wings provide 22% to 28% of its lift in forward flight. In a steep banking turn at slower airspeeds, the low wing can lose lift while it is maintained on the upper wing, resulting in an excessive roll. This is countered by increasing forward airspeed to increase lift on the lower wing. Because of this characteristic, and the aircraft's size and weight, it is not easily maneuverable. Therefore they usually attack in pairs or multiple pairs, and from various directions.

European Utility Helicopter AS-532/COUGAR



Weapon & Ammunition Types	Combat Load
7.65-mm MG	2
Other Loading Options	
20-mm twin gun pods	2
68-mm rocket pods (22 each)	2
2.75-in rocket pods (19 each)	2
External fuel tanks (liters)	600

SYSTEM

Alternative Designations: AS 332 Super Puma, SA 330 Puma
Date of Introduction: 1981
Proliferation: At least 38 countries

Description: Variants in "( )"
Crew: 2 (pilots)
Blades:
  Main rotor: 4
  Tail rotor: 5, 4 (U2/A2)
Engines: 2x 1,877-shp Turbomeca Makila 1A1 turboshaft
Weight (kg):
  Maximum Gross: 9,000 (Mk I), 9,750 (Mk II)
  Normal Takeoff: 8,600 (Mk I), 9,300 (Mk II)
  Empty: 4,330 (UC/AC), 4,460 (UL/AL), 4,760 (U2/A2)
Speed (km/h):
  Maximum (level): 275 (Mk I), 325 (Mk II)
  Cruise: 270
Ceiling (m):
  Service: 4,100
  Hover (out of ground effect): 1,650 (Mk I), 1,900 (Mk II)
  Hover (in ground effect): 2,800 (Mk I), 2,540 (Mk II)
Vertical Climb Rate (m/s): 7
Fuel (liters):
  Internal: 1,497 (UC/AC), 2,000 (UL/AL), 2,020 (U2/A2)
  Internal Aux Tank: 475 ea. (4x Mk I, 5x Mk II)
Range (km):
  Normal Load: 620 (UC/AC), 840 (UL/AL), 800 (U2/A2)
  With Aux Fuel: 1,017 (UC/AC), 1, 245 (UL/AL), 1,176 (U2/A2)
Dimensions (m):
  Length (rotors turning): 18.7-19.5 (U2/A2)

Dimensions continued (m):
  Length (fuselage): 15.5 (UC/AC), 16.3 (UL/AL), 16.8 (U2/A2)
  Width: 3.6-3.8 (U2/A2)
  Height: 4.6
  Main Rotor Diameter: 15.6-16.2 (U2/A2)
  Tail Rotor Diameter: 3.1-3.2 (U2/A2)
Cargo Compartment Dimensions (m):
  Floor Length: 6.5 (AC/UC), 6.8 (UL/AL), 7.9 (U2/A2)
  Width: 1.8
  Height: 1.5
Standard Payload (kg):
  Internal load: 3,000
  External on sling only: 4,500
  Transports 20-29 troops or 6-12 litters (variant dependant), or cargo.

Survivability/Countermeasures:
Main and tail rotor blades electrically deiced.
A radar warning receiver is standard, while a laser warning receiver, missile launch detector, missile approach detector, infrared jammer, decoy launcher, and flare/chaff dispensers are optionally available.

ARMAMENT

The Mk I variants may employ 2x 7.65-mm machine guns on pintle-mounts in the cabin doors when employed in a transport role.

Most Probable Armament
The armed versions have side-mounted 20-mm machineguns and/or axial pods fitted with 68-mm rocket launchers.

AVIONICS/SENSOR/OPTICS

Night/Weather Capabilities:
The aircraft is NVG compatible, and through its instruments, avionics, full autopilot, and nav computer, is capable of operation in day, night, and instrument meteorological conditions.

VARIANTS

SA 330 Puma: Developed in the late 1960s by Aerospatiale in France. Others were built in the UK, Indonesia, Romania.

AS 332 Super Puma: Differs from the SA 330 Puma through an improved rotor system, upgraded engines, stretched fuselage, and a modified nose shape.
The Cougar name was adopted for all military variants, and in 1990, all Super Puma designations were changed from AS 332 to AS 532 to distinguish between civil and military variants. The "5" denotes military, "A" is armed, "C" is armed-antitank, and "U" is utility. The second letter represents the level of "upgrading".

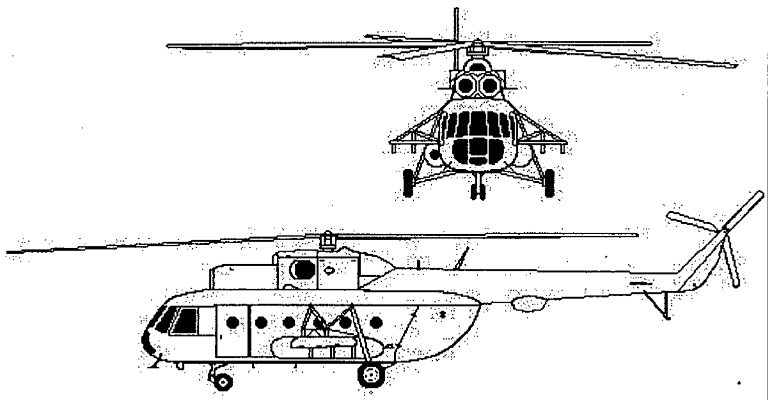
AS-532 Cougar UC/AC Mk I: The basic version with a short fuselage to carry 20 troops.

AS-532 Cougar UL/AL Mk I: This version has an extended fuselage, which allows it to carry 25 troops and more fuel. It is also capable of carrying an external load of 4,500 kg.

AS-532 Cougar U2/A2 Mk II: This 1992 version is the longest variant of the Cougar line. It has an improved Spheriflex rotor system with only 4x tail rotor blades, and 2x 2,100-shp Turbomeca Makila 1A2 turboshaft engines that allow an increased cargo carrying capability. It can transport 29 troops or 12 litters, or an external load of 5,000 kg. Primarily used for combat search and rescue, and as an armed version. It may be armed additionally with a 20-mm cannon or pintle-mounted .50 caliber machine guns.

NOTES
This helicopter is produced by the Eurocopter company. It was formed as a joint venture between Aerospatiale of France, and Daimler-Benz Aerospace of Germany. Additional missions include: VIP transport, electronic warfare, and anti-submarine warfare.

Russian Utility Helicopter Mi-8/HIP



Weapon & Ammunition Types	Combat Load
2x 7.62-mm or 1x 12.7-mm MG	
Other Loading Options	
AT-2C or AT-3 ATGMs	4-6
57-mm rocket pods (16 each)	4-6
80-mm rocket pods (20 each)	2
250-kg bombs	4
500-kg bombs	2
12.7-mm MG pod	2
Twin 23-mm gun pods	2
Additional fuel tanks (liters)	1,830

SYSTEM

Alternative Designations: INA
Date of Introduction: 1967
Proliferation: At least 54 countries

Description:
Crew: 3 (2x pilots, 1x flight engineer)
Blades:
  Main rotor: 5
  Tail rotor: 3
Engines: 2x 1,700-shp Isotov TV2-117A turboshaft
Weight (kg):
  Maximum Gross: 12,000
  Normal Takeoff: 11,100
  Empty: 6,990
Speed (km/h):
  Maximum (level): 250
  Cruise: 225
Ceiling (m):
  Service: 4,500
  Hover (out of ground effect): 800
  Hover (in ground effect): 1,900
Vertical Climb Rate (m/s): 9
Fuel (liters):
  Internal: 445
  Internal Aux Tank: 915 ea.
  External Fuel Tank: 745 in port tank, 680 in starboard tank
Range (km):
  Maximum Load: INA
  Normal Load: 460
  With Aux Fuel: 950

Dimensions (m):
  Length (rotors turning): 25.2
  Length (fuselage): 18.2
  Width: 2.5
  Height: 5.6
  Main Rotor Diameter: 21.3
  Tail Rotor Diameter: 3.9
Cargo Compartment Dimensions (m):
  Floor Length: 5.3
  Width: 2.3
  Height: 1.8
Standard Payload:
  HIP C: 24 troops, or 3,000 kg internal or external loads on 4x hardpoints.
  HIP E: 24 troops, or 4,000 kg internal or 3,000 kg external on 6x hardpoints.
  HIP J/K: antennas on aft section of fuselage.

Survivability/Countermeasures:
Main and tail rotor blades electrically deiced.
Infrared jammer, chaff and flares.

ARMAMENT

Loaded combat troops can fire personal weapons through windows from inside cabin.
The HIP E mounts a flexible 12.7-mm machinegun in the nose.

AVIONICS/SENSOR/OPTICS

Night/Weather Capabilities:
The Mi-8 is equipped with instruments and avionics allowing operation in day, night, and instrument meteorological conditions.

VARIANTS

Mi-8T: The HIP C is a medium assault/ transport version. The probable armament is 57-mm rockets, bombs, or AT-2C/ SWATTER ATGMs.

Mi-8VPK: The HIP D is an airborne communications platform with rectangular communication canisters mounted on weapons racks.

Mi-8TVK: The HIP E is used as a gunship or direct air support platform. Airframe modifications add 2x external hardpoints for a total of 6, and mount a flexible 12.7-mm machinegun in the nose. The probable armament is 57-mm rockets, bombs, or AT-2/SWATTER ATGMs.

Mi-8MT/MTV/MTB/-171-17: The HIP H is an upgraded medium assault/ transport version. See separate Mi-17 entry.

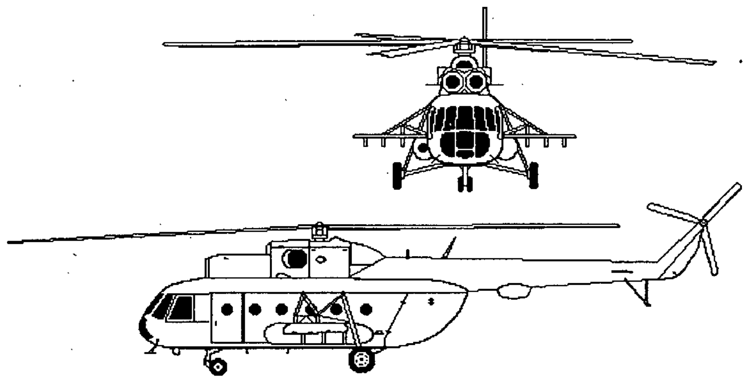
Mi-8SMV: The HIP J is an airborne jamming platform characterized by small boxes on the left side of the fuselage.

Mi-8PPA: The HIP K is an airborne jamming platform characterized by 6x "X"-shaped antennas on the aft fuselage.

Mi-9: The HIP G is an airborne command post characterized by antennas, and Doppler radar on tailboom.

NOTES
Available munitions are shown above; not all may be employed at one time, mission dictates weapon configuration. External stores are mounted on weapons racks on each side of the fuselage. The HIP C has four external hardpoints; the HIP E, HIP H, have six; other variants have none. Interior seats are removable for cargo carrying. The rear clamshell doors open, an internal winch facilitates loading of heavy freight. Floor has tiedown rings throughout. The aircraft carries a rescue hoist capable to 150 kg, and a cargo sling system capable to 3,000 kg. The Mi-8 is capable of single-engine flight in the event of loss of power by one engine (depending on aircraft mission weight) because of an engine load sharing system. If one engine fails, the other engine's output is automatically increased to allow continued flight. See also Mi-17.

Russian Utility Helicopter Mi-17/HIP



Weapon & Ammunition Types	Combat Load
2x 7.62-mm or 1x 12.7-mm MG	
Other Loading Options	
AT-2C or AT-3 ATGMs	4-6
57-mm rocket pods (16 each)	4-6
80-mm rocket pods (20 each)	2
250-kg bombs	4
500-kg bombs	2
12.7-mm MG pod	
Twin 23-mm gun pods	2
Additional fuel tanks (liters)	1,830

SYSTEM

Alternative Designations: Mi-8MT HIP H
Date of Introduction: 1981 (as Mi-17)
Proliferation: At least 22 countries

Description:
Crew: 3 (2x pilots, 1x flight engineer)
Blades:
  Main rotor: 5
  Tail rotor: 3
Engines: 2x 1,950-shp Isotov TV3-117MT turboshaft
Weight (kg):
  Maximum Gross: 13,000
  Normal Takeoff: 11,100
  Empty: 7,100-7,370 (variant dependant)
Speed (km/h):
  Maximum (level): 250
  Cruise: 240
Ceiling (m):
  Service: 5,000-5,700 (variant dependant)
  Hover (out of ground effect): 1,760
  Hover (in ground effect): 1,900-3,980 (variant dependant)
Vertical Climb Rate (m/s): 9
Fuel (liters):
  Internal: 445
  Internal Aux Tank: 915 ea.
  External Fuel Tank:
    Port Tank: 745
    Starboard Tank: 680
Range (km):
  Normal Load: 495
  With Aux Fuel: 1,065

Dimensions (m):
  Length (rotors turning): 25.4
  Length (fuselage): 18.4
  Width: 2.5
  Height: 5.7
  Main Rotor Diameter: 21.3
  Tail Rotor Diameter: 3.9
Cargo Compartment Dimensions (m):
  Floor Length: 5.3
  Width: 2.3
  Height: 1.8
Standard Payload (kg):
  Internal load: 4,000
  External on sling only: 3,000
  Transports 24 troops and cargo, or armaments on 6x external hardpoints.

Survivability/Countermeasures:
Main and tail rotor blades electrically deiced. Infrared jammer, chaff and flares.

ARMAMENT

Loaded combat troops can fire personal weapons through cabin windows from inside cabin.

Most Probable Armament:
HIP H: Fitted with 2x 7.62-mm machineguns or possibly 2x 23-mm GSh-23 gun packs in cabin, 57-mm rockets, and AT3/SAGGER ATGMs.

AVIONICS/SENSOR/OPTICS

Night/Weather Capabilities:
The Mi-17 is equipped with instruments, avionics, Doppler radar, and a fully functioning autopilot for operation in day, night, and instrument meteorological conditions.

VARIANTS

Mi-17: A mid-life upgrade of the widely proliferated Mi-8 HIP H medium assault/ transport helicopter. Initially, only the export version was known as the Mi-17. The only visible differences between this variant and the older Mi-8s is that the tail rotor is on the portside rather than the starboard side, and crew armor plating.

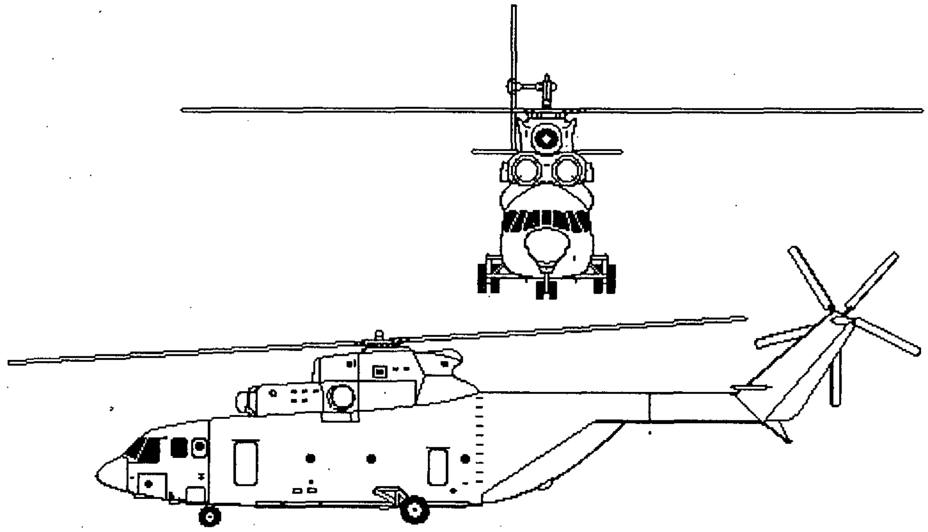
Mi-17P: A descendent of the HIP K airborne jamming platform characterized by large rectangular antennas along the aft fuselage.

Mi-171/-17M/-17V: Also known as **Mi-8MTV**, and a descendent of the HIP H. The engines are upgraded to 2x 2,070-shp Klimov TV3-117VMAs to allow greater rates of climb and hover ceilings, yet performance characteristics remain virtually unchanged from the baseline Mi-17.

Mi-8: See separate entry.

NOTES
Available munitions are shown above; not all may be employed at one time, mission dictates weapon configuration. External stores are mounted on weapons racks on each side of the fuselage. The Mi-17 has six external hardpoints. Additional missions include; attack, direct air support, electronic warfare, airborne early warning, medevac, search and rescue, and minelaying. Interior seats are removable for cargo carrying. The rear clamshell doors open, an internal winch facilitates loading of heavy freight. Floor has tiedown rings throughout. The aircraft carries a rescue hoist capable to 150 kg. The Mi-17 is capable of single-engine flight in the event of loss of power by one engine (depending on aircraft mission weight) because of an engine load sharing system. If one engine fails, the other engine's output is automatically increased to allow continued flight. See also Mi-8.

Russian Transport Helicopter Mi-26/HALO



SYSTEM

Alternative Designations: INA
Date of Introduction: 1983
Proliferation: At least 5 countries

Description:
Crew: 5 (2x pilots, 1x navigator, 1x flight engineer, 1x loadmaster)
Blades:
  Main rotor: 8
  Tail rotor: 5
Engines: 2x 11,400-shp Lotarev D-136 turboshaft
Weight (kg):
  Maximum Gross: 56,000
  Normal Takeoff: 49,500
  Empty: 28,240
Speed (km/h):
  Maximum (level): 295
  Cruise: 255
Ceiling (m):
  Service: 4,500
  Hover (out of ground effect): 1,800
  Hover (in ground effect): 4,500
Vertical Climb Rate: INA

Fuel (liters):
  Internal: 11,900
Range (km):
  Maximum Load: 800
  Normal Load: INA
  With Aux Fuel: 1200 km
Dimensions (m):
  Length (rotors turning): 40
  Length (fuselage): 33.5
  Width: 8.2
  Height: 8.1
  Main Rotor Diameter: 32
  Tail Rotor Diameter: 7.6
Cargo Compartment Dimensions (m):
  Floor Length: 12
  Width: 3.3
  Height: variable from 2.9 to 3.2
Standard Payload:
  Internal or external load: 20,000 kg
  Transports over 80 troops, 60 litters, or 2x BRDM-2 scout cars, or 2x BMDs, or 1x BMP or, 1x BTR-60/70/80 or, 1x MT-LB.

Survivability/Countermeasures:
Main and tail rotor blades electrically deiced.
Infrared signature suppressors on engines.
Infrared jammers and decoys; flares.
Self-sealing fuel tanks.

ARMAMENT

None

AVIONICS/SENSOR/OPTICS

Night/Weather Capabilities:
The avionics and navigational package, a Doppler weather radar, and a fully functioning autopilot allow for day/night all-weather operation.

VARIANTS

Mi-26MS: Medical evacuation version.

Mi0-26T: Freight transport.

Mi-26TZ: Fuel tanker with an additional 14,040 liters of fuel in 4x internal tanks and 1,040 liters of lubricants, pumped through 4x 60-meter long refueling nozzles for refueling aircraft, and 10x 20-meter long hoses for refueling ground vehicles. Fuel transfer rate is 300 liters/minute for aviation fuel, and 75-150 liters/minute for diesel fuel. The refueling system can easily be removed to allow the aircraft to perform transport missions.

NOTES
The HALO A has no armament. The load and lift capabilities of the aircraft are comparable to the U.S. C-130 Hercules transport aircraft. The length of the landing gear struts can be hydraulically adjusted to facilitate loading through the rear doors. The tailskid is retractable to allow unrestricted approach to the rear clamshell doors and loading ramp. The cargo compartment has two electric winches (each with 2,500 kg capacity) on overhead rails can move loads along the length of the cabin. The cabin floor has rollers and tie-down rings throughout. The HALO has a closed-circuit television system to observe positioning over a sling load, and load operations. The Mi-26 is capable of single-engine flight in the event of loss of power by one engine (depending on aircraft mission weight) because of an engine load sharing system. If one engine fails, the other engine's output is automatically increased to allow continued flight.

Glossary

AA - antiaircraft
acquisition range - sensor range against a category of targets. Targets are usually categorized as infantry, armored vehicles, or aircraft. Acquisition includes four types (or levels of clarity, in ascending order of clarity): detection, classification, recognition, and identification. Where the type of acquisition is not specified, the acquisition range will be regarded as sufficient for accurate targeting. This range is comparable to the former Soviet term *sighting range*.
AAM - air-to-air missile
AGL - automatic grenade launcher
AIFV- airborne infantry fighting vehicle
aka - also known as
antitank - functional area and class of weapons characterized by destruction of tanks. In the modern context, used in this guide, the role has expanded to the larger one of "antiarmor". Systems and munitions employed against light armored vehicles may be included within the category of antitank.
AP - antipersonnel
APE - armor-piercing explosive (ammunition)
APC - armored personnel carrier
APC-T - armor-piercing capped tracer (ammunition)
AP HE - armor-piercing high explosive (ammunition)
API-T - armor-piercing incendiary tracer (ammunition)
APERS-T - antipersonnel tracer (ammunition)
APT - armor-piercing tracer (ammunition)
APU - auxiliary power unit; auxiliary propulsion unit
ASM - air-to-surface missile
AT - antitank
ATGL - antitank grenade launcher
ATGM - antitank guided missile
average cross-country (speed) - vehicle speed (km/hr) on unimproved terrain without a road.

burst (rate of fire) - artillery term: the greatest number of rounds that can be fired in 1 minute.

caliber - munition diameter (mm or inches), used to classify munition sizes; barrel length of a cannon (howitzer or gun), measured from the face of the breech recess to the muzzle.
canister - close-range direct-fire ammunition which dispenses a fan of flechettes forward
CC - cargo-carrying (ammunition)
CCM - counter-countermeasure
CE - chemical energy: the class of ammunition which employs a shaped charge for the lethal mechanism. Ammunition types which employ CE include HEAT and HESH (see below).
CM - countermeasure
coax - coaxial
CRV - combat reconnaissance vehicle
cyclic (rate of fire) - maximum rate of fire for an automatic weapon (in rd/min)

decon - decontamination
direct-fire range - maximum range of a weapon, operated in the direct-fire mode, at which the bullet's trajectory will not rise above the height of the intended point of impact on the target. At this range, the gunner is not required to adjust for range in order to aim the weapon. The comparable Russian term is *point blank range*.
DPICM - dual-purpose improved conventional munitions (ammunition)
DPICM-BB - dual-purpose improved conventional munitions, base-bleed (ammunition)
DU - depleted uranium (ammunition)
DVO - direct-view optics
ECM - electronic countermeasures
EO - electro-optic, electro-optical
ERA - explosive reactive armor
ERFB - extended range full-bore (ammunition)
ERFB-BB - extended range full-bore, base-bleed (ammunition)

est - estimate
ET - electronic timing (ammunition fuze type)
European - from a consortium of firms located or headquartered in several European countries

FAE - fuel-air explosive (ammunition). This munition technology is employed in aerial bombs and artillery munitions, and uses a dispersing explosive fill to produce intense heat, a long-duration high-pressure wave, and increased HE blast area
FCS - fire control system
FFAR - folding-fin aerial rockets
flechette - former-Soviet artillery ammunition which dispenses flechettes forward over a wide area. Unlike **canister rounds,** these rounds use a time fuze, which permits close-in direct fire, long-range direct fire, and indirect fire.
FLIR - forward-looking infrared (thermal sensor)
FLOT - forward line of own troops
FOV - field of view
frag-HE - fragmentation-high explosive (ammunition)
FSU - former Soviet Union

gen - generation. Equipment such as APS and (thermal and II) night sights are often categorized in terms of 1st, 2nd or 3rd generation of development, with different capabilities for each.
GP MG - general purpose machinegun
GPS - global positioning system

HE - high explosive (ammunition)
HEAT - high-explosive antitank (also referred to as shaped-charge ammunition)
HEAT-FS - high-explosive antitank, fin-stabilized (ammunition)
HEAT-MP - high-explosive antitank, multi-purpose
HEFI - high-explosive fragmentation incendiary (ammunition)
HEI - high-explosive incendiary (ammunition)
HEP-T - high explosive plastic-tracer (ammunition)

HESH - high-explosive squash head (ammunition)
HUD - head-up display
HVAP-T - hypervelocity, armor-piercing tracer (ammunition)

I-T - incendiary - tracer (ammunition)
IFF - identification friend-or-foe
IFV - infantry fighting vehicle
II - image intensification (night sighting system)
ILS - instrument landing system
INA - information not available
IR - infrared

K-kill - catastrophic kill (simulation lethality data)
KE - kinetic energy: class of ammunition which transfers energy to the target for the lethal mechanism. Ammunition types which employ KE include AP, APFSDS-T, and HVAP-T.

LAFV - light armored fighting vehicle
LLLTV - low-light-level television
LMG - light machinegun
LRF - laser rangefinder

max - maximum
maximum aimed range - maximum range of a weapon (based on firing system, mount, and sights) for a given round of ammunition, while aiming at a ground target or target set with sights in the direct-fire mode. The range is not based on single-shot hit probability on a point target, rather on tactical guidance for firing multiple rounds if necessary to achieve a desired lethality effect. One writer referred to this as *range with the direct laying sight*. Even greater ranges were cited for *salvo fire*, wherein multiple weapons (e.g., tank platoon) will fire a salvo against a point target.
max effective range - maximum range at which a weapon may be expected to achieve a high single-shot probability of hit (50%) and a required level of destruction against assigned targets. This figure may vary for each specific munition and by type of target (such as infantry, armored vehicles, or aircraft).
max off-road (speed) - vehicle speed (km/hr) on dirt roads.
MCLOS - manual command-to-line-of-sight
MG - machinegun
Mk - Mark
MRL - multiple rocket launcher

N/A - not applicable
NBC - nuclear, biological, and chemical
Nd - neodymium, type of laser rangefinder
NFI - no further information
normal (rate of fire) - artillery term: rate (in rd/min) for fires over a 5-minute period.
NVG - night-vision goggle
NVS - night-vision system

PD - point-detonating (ammunition fuze type)
Ph - probability of hit (simulation lethality data)
PIBD - point-initiating base-detonating (ammunition fuze type)
pintel - post attached to a firing point or vehicle, used to replace the base for a weapon mount
Pk - probability of kill (simulation lethality data)
practical (rate of fire) - maximum rate of fire for sustained aimed weapon fire against point targets. The rate includes reload time and reduced rate to avoid damage from overuse. Former Soviet writings also refer to this as the **technical rate of fire**.

recon - reconnaissance
Rd - round
ready rounds - rounds available for use on a weapon, whether in autoloader or in nearby stowage, which can be loaded within the weapon's stated rate of fire.
RF - radio frequency
RHA - rolled homogeneous armor, often used as a standard armor hardness for measuring penetration of anti-tank munitions.
RHAe - RHA equivalent, a standard used for measuring penetrations against various type armors

SACLOS - semiautomatic command-to-line-of-sight
SAM - surface-to-air missile
SP - self-propelled
stadiametric - in this guide, a method of range-finding using stadia line intervals in sights and target size within those lines to estimate target range.
stowed rounds - rounds available for use on a weapon, but stowed and requiring a delay greater than that for ready rounds (and cannot be loaded within the weapon's stated rate of fire).
sustained (rate of fire) - artillery term: rate (in rd/min) for fires over the duration of an hour.

tactical AA range - maximum targeting range against aerial targets, aka: **slant range**.
TAR - target acquisition radar
TELAR - transporter-erector-launcher and radar
thermobaric - HEI volumetric (blast effect) explosive technology similar to fuel-air explosive and used in shoulder-fired infantry weapons and ATGMs.
TLAR - transporter-launcher and radar
TOF - time of flight (seconds)
TTP - tactics, techniques, and procedures
TTR - target tracking radar

UI - unidentified

VEESS - vehicle engine exhaust smoke system
vs - versus

w/ - with (followed by associated item)
WP - white phosphorus (ammunition)

EXHIBIT PP

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE SECRETARY

In the Matter of	)	
Pa'ina Hawaii, LLC	)	Docket No. 030-36974
	)	
Materials License Application	)	
	)	

DECLARATION OF DR. GORDON R. THOMPSON IN SUPPORT OF PETITIONER'S AREAS OF CONCERN

I, Gordon R. Thompson, declare that if called as a witness in this action I could testify of my own personal knowledge as follows:

I. INTRODUCTION

I-1. I am the executive director of the Institute for Resource and Security Studies (IRSS), a nonprofit, tax-exempt corporation based in Massachusetts. Our office is located at 27 Ellsworth Avenue, Cambridge, Massachusetts 02139. IRSS was founded in 1984 to conduct technical and policy analysis and public education, with the objective of promoting peace and international security, efficient use of natural resources, and protection of the environment. In addition to holding my position at IRSS, I am also a research professor at the George Perkins Marsh Institute, Clark University, Worcester, Massachusetts. My professional qualifications are discussed in Section II of this declaration.

I-2. I have been retained by Concerned Citizens of Honolulu as an expert witness in a proceeding before the US Nuclear Regulatory Commission (NRC), regarding an application by

Pa'ina Hawaii, LLC, for a license to build and operate a commercial pool-type industrial irradiator in Honolulu, Hawai'i, at the Honolulu International Airport.

I-3. The purpose of this declaration is to support Concerned Citizens' contention that "special circumstances" exist, precluding the NRC's use of a categorical exclusion from the National Environmental Policy Act's mandate to prepare either an environmental assessment (EA) or environmental impact statement (EIS) in the context of the proposed license.[1] In this declaration, I focus on the potential for acts of malice or insanity, related to the proposed Pa'ina Hawaii irradiator, to cause harm to people and/or the environment. As part of that focus, I address the potential to reduce the risk of harm by adopting alternatives to the proposed mode of construction and operation of the irradiator. Also, I address the processes whereby acts of malice or insanity could be considered in a licensing proceeding or during the preparation of an EA or EIS. My focus on the implications of potential acts of malice or insanity does not indicate that I regard other issues, relevant to licensing of the proposed irradiator, as having a lesser significance.

I-4. The remainder of this declaration has seven sections. Section II discusses my professional qualifications. Section III discusses some of the characteristics of the proposed Pa'ina Hawaii irradiator. The potential for commercial nuclear facilities, including irradiators, to be affected by acts of malice or insanity is addressed in Section IV. That discussion is continued in Section V, with a focus on irradiators. Section VI discusses the potential to reduce the risk of harm, arising from acts of malice or insanity, by adopting alternatives to the proposed design and mode of operation of the Pa'ina Hawaii irradiator. Section VII addresses the processes whereby acts of malice or insanity could be considered in a licensing proceeding, or during the

[1] 10 C.F.R. § 51.22(b); see also id. § 2.335(b); 40 C.F.R. § 1508.4.

preparation of an EA or EIS, for the Pa'ina Hawaii irradiator. Major conclusions are set forth in Section VIII. Documents cited in this declaration are listed in a bibliography that is appended to the declaration.

II. MY PROFESSIONAL QUALIFICATIONS

II-1. I received an undergraduate education in science and mechanical engineering at the University of New South Wales, in Australia. Subsequently, I pursued graduate studies at Oxford University and received from that institution a Doctorate of Philosophy in mathematics in 1973, for analyses of plasmas undergoing thermonuclear fusion. During my graduate studies I was associated with the fusion research program of the UK Atomic Energy Authority. My undergraduate and graduate work provided me with a rigorous education in the methodologies and disciplines of science, mathematics, and engineering.

II-2. Since 1977, a significant part of my work has consisted of technical analyses of safety, security and environmental issues related to nuclear facilities. These analyses have been sponsored by a variety of nongovernmental organizations and local, state and national governments, predominantly in North America and Western Europe. Drawing upon these analyses, I have provided expert testimony in legal and regulatory proceedings, and have served on committees advising US government agencies. In a number of instances, my technical findings have been accepted or adopted by relevant governmental agencies. To illustrate my expertise, I provide in the following paragraphs some details of my experience.

II-3. During the period 1978-1979, I served on an international review group commissioned by the government of Lower Saxony (a state in Germany) to evaluate a proposal for a nuclear fuel cycle center at Gorleben. I led the subgroup that examined safety and security risks, and identified alternative options with lower risk. One of the risk issues that I identified and analyzed was the potential for self-sustaining, exothermic oxidation reactions of fuel cladding in a high-density spent-fuel pool if water is lost from the pool. Hereafter, for simplicity, this event is referred to as a "pool fire". In examining the potential for a pool fire, I identified partial loss of water as a more severe condition than total loss of water. I identified a variety of events that could cause a loss of water from a pool, including aircraft crash, sabotage, terrorism and acts of war. Also, I identified and described alternative spent-fuel-storage options with lower risk; these lower-risk options included design features such as spatial separation, natural cooling and underground vaults. The Lower Saxony government accepted my findings about the risk of a pool fire, and ruled in May 1979 that high-density pool storage of spent fuel was not an acceptable option at Gorleben. As a direct result, policy throughout Germany has been to use dry storage in casks, rather than high-density pool storage, for away-from-reactor storage of spent fuel.

II-4. My work has influenced decision making by safety officials in the US Department of Energy (DOE). During the period 1986-1991, I was commissioned by environmental groups to assess the safety of the military production reactors at the Savannah River Site, and to identify and assess alternative options for the production of tritium for the US nuclear arsenal. Initially, much of the relevant information was classified or otherwise inaccessible to the public. Nevertheless, I addressed safety issues through analyses that were recognized as accurate by nuclear safety officials at DOE. I eventually concluded that the Savannah River reactors could not meet the safety objectives set for them by DOE. The Department subsequently reached the same conclusion, and scrapped the reactors. Current national policy for tritium production is to employ commercial reactors, an option that I had concluded was technically attractive but problematic from the perspective of nuclear weapons proliferation.

II-5. In 1977, and again during the period 1996-2000, I examined the safety and security of nuclear fuel reprocessing and liquid high-level radioactive waste management facilities at the Sellafield site in the UK. My investigation in the latter period was supported by consortia of local governments in Ireland and the UK, and I presented findings at briefings in the UK and Irish parliaments in 1998. I identified safety issues that were not addressed in any publicly available literature about the Sellafield site. As a direct result of my investigation, the UK Nuclear Installations Inspectorate (NII) required the operator of the Sellafield site -- British Nuclear Fuels -- to conduct extensive safety analyses. These analyses confirmed the significance of the safety issues that I had identified, and in January 2001 the NII established a legally binding schedule for reduction of the inventory of liquid high-level radioactive waste at Sellafield. The NII took this action in recognition of the grave offsite consequences of a release to the environment from the tanks in which liquid high-level waste is stored. I had identified a variety of events that could cause such a release, including acts of malice or insanity.

II-6. In January 2002, I authored a submission to the UK House of Commons Defence Committee, addressing the potential for civilian nuclear facilities to be used by an enemy as radiological weapons. The submission drew upon my own work, and the findings of other analysts, dating back as far as the mid-1970s. My primary recommendation was that the Defence Committee should call upon the Parliamentary Office of Science and Technology (POST) to conduct a thorough, independent analysis of this threat. I argued that the UK government and nuclear industry could not be trusted to provide a credible analysis. The Defence Committee subsequently adopted my recommendation, and a study was conducted by POST.

II-7. I was the author or a co-author of two documents, published in 2003, that addressed the safety and security risks arising from the storage of spent fuel in high-density pools at US nuclear power plants.[2] This work expanded on analysis that I had first conducted in the context of the proposed nuclear fuel cycle center at Gorleben, as discussed in paragraph II-3, above. The two documents became controversial, and their findings and recommendations were challenged by the NRC. The US Congress recognized that our findings, if correct, would be significant for national security. Accordingly, Congress requested the National Academy of Sciences (NAS) to conduct an independent investigation of these issues. The Academy's report vindicated the work done by my co-authors and me.[3]

III. CHARACTERISTICS OF THE PROPOSED IRRADIATOR

III-1. According to the NRC, Pa'ina Hawaii has stated that the proposed irradiator would be used primarily for the irradiation of fresh fruit and vegetables bound for the US mainland. Other items to be irradiated would include cosmetics and pharmaceutical products.[4] A story in the technical press has stated that the irradiator would be the Genesis model manufactured by Gray-Star, using a 1 million-Curie Cobalt-60 source located in a water-filled pool 22 feet deep.[5] Cobalt-60 is a radioactive isotope with a half-life of 5.3 years. According to an April 2004 NRC fact sheet, all US commercial irradiators regulated by the NRC currently use Cobalt-60; the amount used at each irradiator typically exceeds 1 million Curies and can range up to 10 million

[2] Thompson, 2003; Alvarez et al, 2003.
[3] NAS, 2005.
[4] NRC, 2005.
[5] Nuclear News, 2005.

Curies.[6] The Cobalt-60 is present in the form of sealed sources typically consisting of metallic "pencils" said to be about one inch in diameter and one foot long.[7]

III-2. The version of Pa'ina Hawaii's license application that has been posted at the NRC website has major redactions. That document does not allow the reaching of any conclusion about the safety and security of the proposed irradiator.

IV. THE POTENTIAL FOR NUCLEAR FACILITIES TO BE AFFECTED BY ACTS OF MALICE OR INSANITY

IV-1. No commercial nuclear facility in the United States was designed to resist attack. Facilities have some capability in this respect by virtue of design for other objectives (e.g., resisting tornado-driven missiles). Beginning in 1994, with the NRC's promulgation of a vehicle-bomb rule, each US nuclear power plant has implemented site-security measures (e.g., barriers, guards) that have some capability to prevent attackers from damaging vulnerable parts of the plant. The scope of this defense was increased in response to the attacks of 11 September 2001. Nevertheless, it continues to reflect the NRC's judgment that a "light defense" of nuclear power plants, to use military terminology, is sufficient.[8] This judgment is not supported by any published strategic analysis. The NRC takes the same approach in regulating nuclear facilities other than power plants, including commercial irradiators.

IV-2. A strategic analysis of needs and opportunities for security of a nuclear facility should have three parts. It should begin with an assessment of the scale of damage that could arise from an attack. A major determinant of this scale is the amount of radioactive material that is available for release to the atmosphere or a water body; other determinants are the

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[6] NRC, 2004b.
[7] Kelly, 2002.
[8] NRC, 2004a.

vulnerability of the facility to attack, and the consequences of attack.[9] The second step in the strategic analysis should be to assess the future threat environment. The third step should be to assess the adequacy of present measures to defend the facility, and to identify options for providing an enhanced defense.

IV-3. The analyst should seek to understand the interests and perspectives of potential attackers. To illustrate, a sub-national group that is a committed enemy of the United States might perceive two major incentives for attacking a US commercial nuclear facility. First, release of a large amount of radioactive material could cause major, lasting damage to the United States. Second, commercial nuclear technology could symbolize US military dominance through nuclear weapons and associated technologies such as guided missiles; a successful attack on a commercial nuclear facility could challenge that symbolism. Conversely, the group might perceive three major disincentives for attack. First, nuclear facilities could be less vulnerable than other potential targets. Second, radiological damage from the attack would be indiscriminate, and could occur hundreds of km downwind in non-enemy locations (e.g., Mexico). Third, the United States could react with extreme violence.

IV-4. The threat environment must be assessed over the entire period during which a nuclear facility is expected to operate. For spent-fuel storage facilities, that period could exceed a century. The risk of attack will accumulate over the period of operation. Forecasting international conditions over several decades is a notoriously difficult and uncertain enterprise. Nevertheless, an implicit or explicit forecast must underlie any decision about the level of security that is provided at a nuclear facility. Prudence dictates that a forecast in this context

[9] Direct release of radioactive material is not the only potential consequence of an attack on a nuclear facility. There is also concern that radioactive or fissile material could be removed from the facility and incorporated into a radiological or nuclear weapon.

should err on the side of pessimism. Decision makers should, therefore, be aware of a literature indicating that the coming decades could be turbulent, with a potential for higher levels of violence.[10] One factor that might promote violence is a perception of resource scarcity. It is noteworthy that many analysts are predicting a peak in world oil production within the next few decades.[11] Also, a recent international survey shows significant degradation in the Earth's ability to provide ecosystem services.[12]

IV-5. The potential for attacks on nuclear facilities has been studied for decades.[13] Nevertheless, the NRC remains convinced that these facilities require only a light defense. The NRC's position fails to account for the growing strategic significance of sub-national groups as potential enemies. Various groups of this kind could possess the motive and ability to mount an attack on a US nuclear facility with a substantial probability of success. The unparalleled military capability of the United States cannot deter such a threat if the attacking group has no territory that could be counter-attacked. Moreover, use of US military capability could be counter-productive, creating enemies faster than they are killed or captured. Many analysts believe that the invasion of Iraq has produced that outcome.

IV-6. The discussion in the preceding paragraphs shows that it would be prudent to consider options for providing an enhanced defense of nuclear facilities. Design studies have identified a large potential for increasing the robustness of new facilities.[14] This finding argues for careful consideration of alternative options during the licensing of a new facility. At existing facilities, there is usually less opportunity for increasing robustness. Nevertheless, there are

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[10] Kugler, 1995; Raskin et al, 2002.
[11] Hirsch et al, 2005.
[12] Stokstad, 2005.
[13] Ramberg, 1984.
[14] Hannerz, 1983.

many opportunities to enhance the defenses of an existing facility. I have identified such opportunities in a number of instances. For example, I have identified a set of measures that could provide an enhanced defense of the San Onofre nuclear power plant.[15]

V. POTENTIAL ACTS OF MALICE OR INSANITY IN THE CONTEXT OF IRRADIATORS

V-1. Section IV, above, shows that it would be prudent, in the licensing and regulation of a range of nuclear facilities, to consider the implications of potential acts of malice or insanity. Commercial irradiators, such as that proposed by Pa'ina Hawaii, are among the facilities for which this consideration would be prudent. The reason is that these irradiators contain large amounts of Cobalt-60. If that material were removed from its containment and brought into proximity to humans and other life forms or their habitats, significant harm could occur. The nature of that harm is illustrated by a case study that is discussed in paragraph V-3, below.

V-2. An act of malice or insanity could remove Cobalt-60 from its containment, and bring this material into potential proximity to life forms, in two ways. First, a violent event involving mechanisms such as blast, impact and fire could release Cobalt-60 to the atmosphere from the irradiator facility or during transport of Cobalt-60 sealed sources to or from the facility.[16] This violent event could be a deliberate attack or, conceivably, a collateral event deriving from an attack directed elsewhere. Second, Cobalt-60 sealed sources could be removed intact from the irradiator facility or during transport to or from the facility, and these sources could be used to deliberately irradiate life forms or their habitats. This irradiation could be accomplished by atmospheric dispersal of Cobalt-60 from a sealed source, with or without

[15] Thompson, 2004.

[16] After release to the atmosphere, the Cobalt-60 would be present in fragments or particles of various sizes, which would eventually be deposited on the ground around or downwind of the point of release.

chemical and physical manipulation of the source prior to dispersal.[17] An explosive charge could be used to achieve dispersal, a process that is commonly described as the use of a "dirty bomb". Atmospheric dispersal might also be achieved, after chemical and physical manipulation of the source, through mechanisms such as spraying and combustion. As an alternative to atmospheric dispersal, hostile irradiation could be accomplished by clandestinely placing sealed sources, or fragments thereof, in locations (e.g., bus or train stations) where targeted populations are likely to be present.[18]

V-3. Findings of a theoretical case study on atmospheric dispersal of Cobalt-60 were summarized in Congressional testimony by the Federation of American Scientists in 2002.[19] The case study assumed that one Cobalt-60 "pencil" from a commercial irradiator would be explosively dispersed at the lower tip of Manhattan. The results were compared with those from an assumed dispersal of radioactive cesium, in the following statement:[20]

> "Again, no immediate evacuation would be necessary, but in this case [the Cobalt-60 dispersal], an area of approximately one thousand square kilometers, extending over three states, would be contaminated. Over an area of about three hundred typical city blocks, there would be a one-in-ten risk of death from cancer for residents living in the contaminated area for forty years. The entire borough of Manhattan would be so contaminated that anyone living there would have a one-in-a-hundred chance of dying from cancer caused by the residual radiation. It would be decades before the city was inhabitable again, and demolition might be necessary."

V-4. Following an atmospheric dispersal of radioactive material such as Cobalt-60, the area of land that would be regarded as contaminated, and the overall economic consequences of the event, would depend on the contamination standard that would apply.[21] At present, there are

[17] Zimmerman and Loeb, 2004.
[18] NRC, 2003.
[19] Kelly, 2002.
[20] Kelly, 2002.
[21] Reichmuth et al, 2005.

competing standards, and no clarity about which one would apply.[22] Resolving this issue could be politically difficult, either before or after a dispersal event. A further complicating factor is the exclusion of radiation risk from virtually all insurance policies written in the United States.[23]

V-5. A malicious actor who seeks to expose a population to radioactive material, such as Cobalt-60, could have a range of goals including: (i) causing prompt casualties; (ii) spreading panic; (iii) recruitment to the actor's cause; (iv) asset denial; (v) economic disruption; and (vi) causing long-term casualties.[24]

V-6. Many public officials in the United States and elsewhere are aware of the threat of malicious exposure to radioactive material. At times, substantial resources have been allocated to addressing this threat. For example, a major US government effort was mounted in December 2003 to detect "dirty bombs" in various US cities.[25] Recently, the Australian government has located large, unsecured radioactive sources in two countries in Southeast Asia. At least one of these sources was Cobalt-60.[26] Acting in a manner that invites comparison with licensing of the proposed Pa'ina Hawaii irradiator, the National Nuclear Security Administration (NNSA) removed Cobalt-60 from an irradiator at the University of Hawai'i in March 2005.[27] This removal occurred during the same week in which the NRC issued a Notice of Violation that responded to an NRC-observed security breach at the irradiator in March 2003.[28] It is said that

[22] Medalia, 2004; Zimmerman and Loeb, 2004.
[23] Zimmerman and Loeb, 2004.
[24] Medalia, 2004.
[25] Mintz and Schmidt, 2004.
[26] Eccleston and Walters, 2005.
[27] NNSA, 2005.
[28] Environment Hawai'i, 2005b.

the irradiator contained about 1,000 Curies of Cobalt-60.[29] An NNSA official described the removal of this Cobalt-60 as follows:[30]

> "The removal of these radiological sources has greatly reduced the chance that radiological materials could get into the wrong hands. The university of Hawaii, its surrounding neighbors and the international community are safer today as [a] result of this effort."

V-7. There is a comparatively small technical literature on the safety and security of commercial irradiators, although it is known that safety and security incidents have occurred at these facilities.[31] Irradiators represent one application of sealed radioactive sources. Overall, the use of those sources has created grounds for concern from the perspective of security. According to NRC data, there were more than 1,300 instances of lost, stolen and abandoned sealed sources in the United States between 1998 and 2002.[32]

V-8. In June 2003, the NRC issued its first security order requiring enhanced security at large commercial irradiators.[33] The nature and scope of the required security measures have not been publicly disclosed. It is noteworthy that NRC officials have said that the NRC lacks sufficient staff to conduct inspections of all sealed-source licensees that are expected to receive security orders.[34]

V-9. If provided with relevant information about the design of commercial irradiators, and the security measures that are in effect at these facilities, independent analysts could assess the vulnerability of these facilities to potential acts of malice or insanity. That assessment could be performed in a manner such that sensitive information is not publicly disclosed. The

[29] Environment Hawai'i, 2005a.
[30] NNSA, 2005.
[31] NRC, 1983.
[32] GAO, 2003, page 17.
[33] GAO, 2003, page 28.
[34] GAO, 2003, page 31.

assessment could, for example, assess the vulnerability of irradiators to shaped charges.[35] Also, the assessment could examine the NRC's undocumented assertion that it has "preliminarily determined that it would be extremely difficult for someone to explode a cobalt-60 source in a way that could cause widespread contamination".[36] As explained in paragraph V-2, above, explosive dispersal of an intact Cobalt-60 sealed source is one, but not the only, mechanism whereby Cobalt-60 could be brought into proximity to targeted populations.

VI. ALTERNATIVE OPTIONS

VI-1. The currently-proposed design and mode of operation of the Pa'ina Hawaii irradiator implies a risk of harm to people and/or the environment, arising from potential acts of malice or insanity. Assessment of the nature and scale of that risk must await the provision of more information about the facility than is now publicly available. It is, however, already clear that lower-risk options exist. These options could be systematically examined in an EIS.

VI-2. Two options are available that could eliminate the risk. One such option would be to adopt non-irradiative methods of treating fresh fruit and vegetables. The second option would to use an irradiator that does not require radioactive material such as Cobalt-60. In this context, it is noteworthy that an existing commercial irradiator in Hawai'i employs electron-beam technology. This facility, known as Hawai'i Pride, was built at Kea'au in 2000. Some observers question whether two irradiators, or even one, can be economically viable in Hawai'i.[37]

VI-3. If the Pa'ina Hawaii irradiator were to be built and operated, using Cobalt-60, its design, location and mode of operation could be modified to reduce the risk of harm arising from potential acts of malice or insanity. For example, site security and the robustness of the facility

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[35] Walters, 2003.
[36] NRC, 2004b.
[37] Environment Hawai'i, 2005c.

could be enhanced. Alternative locations could potentially reduce the risk in two ways. First, the currently-proposed location might be especially attractive to attackers because of the proximity of military and symbolic targets including Hickam Air Force Base and Pearl Harbor. Second, the currently-proposed location at Honolulu International Airport might facilitate attack from the air by, for example, an explosive-laden general aviation aircraft. Full delineation of potential modifications, and assessment of their costs and contributions to risk reduction, must await the provision of more information about the facility than is now publicly available.

VII. CONSIDERATION OF ACTS OF MALICE OR INSANITY IN A LICENSE PROCEEDING, EA, OR EIS

VII-1. During an open session of a license proceeding, or in the published version of an EA or EIS, it would be inappropriate to disclose information that could assist the perpetrator of an act of malice or insanity that affects a nuclear facility. It does not follow, however, that acts of malice or insanity cannot be considered in a license proceeding, an EA, or an EIS. Well-tested procedures are available whereby this consideration could occur without publicly disclosing sensitive information. In the context of a license proceeding, some of the sessions, and the accompanying documents, could be open only to authorized persons. Similarly, an EA or EIS could contain sections or appendices that are available only to authorized persons. Interested parties, including public-interest groups, could nominate representatives, attorneys and experts who can become authorized persons on their behalf.

VIII. MAJOR CONCLUSIONS

VIII-1. It would be prudent, in the licensing and regulation of a range of nuclear facilities, to consider the implications of potential acts of malice or insanity. Commercial irradiators, such as that proposed by Pa'ina Hawaii, are among the facilities for which this consideration would be prudent.

VIII-2. The currently-proposed design and mode of operation of the Pa'ina Hawaii irradiator implies a risk of harm to people and/or the environment, arising from potential acts of malice or insanity. Assessment of the nature and scale of that risk must await the provision of more information about the facility than is now publicly available. It is, however, already clear that lower-risk options exist. These options could be systematically examined in an EIS.

VIII-3. Well-tested procedures are available whereby acts of malice or insanity could be considered in a license proceeding, an EA, or an EIS related to the proposed Pa'ina Hawaii irradiator.

I declare under penalty of perjury that I have read the foregoing declaration and know the contents thereof to be true of my own knowledge.

Dated at Cambridge, Massachusetts, 3 October 2005.

GORDON R. THOMPSON

APPENDIX: BIBLIOGRAPHY

(Alvarez et al, 2003)
Robert Alvarez, Jan Beyea, Klaus Janberg, Jungmin Kang, Ed Lyman, Allison Macfarlane, Gordon Thompson and Frank N. von Hippel, "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States", *Science and Global Security*, Volume 11, 2003, pp 1-51.

(Eccleston and Walters, 2005)
Roy Eccleston and Patrick Walters, "Fears for 'dirty bomb' in region", *The Australian*, 29 August 2005.

(Environment Hawai'i, 2005a)
"Honolulu Airport Is Proposed as Site For Cobalt-60 Food Irradiation Facility" *Environment Hawai'i*, Volume 16, Number 3, September 2005, pp 1, 4-6.

(Environment Hawai'i, 2005b)
"NRC Sanctions University of Hawai'i For Lax Irradiator Management" *Environment Hawai'i*, Volume 16, Number 3, September 2005, pp 3-4.

(Environment Hawai'i, 2005c)
"Can Hawai'i Support Two Irradiators?" *Environment Hawai'i*, Volume 16, Number 3, September 2005, page 7.

(GAO, 2003)
US General Accounting Office, *Nuclear Security: Federal and State Action Needed to Improve Security of Sealed Radioactive Sources, GAO-03-804* (Washington, DC: General Accounting Office, August 2003).

(Hannerz, 1983)
K. Hannerz, *Towards Intrinsically Safe Light Water Reactors* (Oak Ridge, Tennessee: Institute for Energy Analysis, February 1983).

(Hirsch et al, 2005)
Robert L. Hirsch, Roger H. Bezdek and Robert M. Wendling, "Peaking Oil Production: Sooner Rather Than Later?" *Issues in Science and Technology*, Volume XXI, Number 3, Spring 2005, pp 25-30. (This paper was adapted from a report prepared for the US Department of Energy's National Energy Technology Laboratory.)

(Kelly, 2002)
Henry Kelly, President, Federation of American Scientists, testimony before the US Senate Committee on Foreign Relations, 6 March 2002.

(Kugler, 1995)
Richard L. Kugler, *Toward a Dangerous World: US National Security Strategy for the Coming Turbulence* (Santa Monica, California: RAND, 1995).

(Medalia, 2004)
Jonathan Medalia, *Terrorist "Dirty Bombs": A Brief Primer* (Washington, DC: Congressional Research Service, Library of Congress, 1 April 2004).

(Mintz and Schmidt, 2004)
John Mintz and Susan Schmidt, "'Dirty Bomb' Was Major New Year's Worry", *Washington Post*, 7 January 2004, page A01.

(NAS, 2005)
Committee on the Safety and Security of Commercial Spent Nuclear Fuel Storage, Board on Radioactive Waste Management, National Research Council, *Safety and Security of Commercial Spent Nuclear Fuel Storage: Public Report* (Washington, DC: National Academies Press, 2005).

(NNSA, 2005)
National Nuclear Security Administration, "NNSA Removes Radioactive Sources From University Facility", Press Release NA-05-07, 13 April 2005.

(NRC, 2005)
US Nuclear Regulatory Commission, "NRC Announces Opportunity for Hearing on License Application for Commercial Irradiator in Honolulu, Hawaii", *NRC News, No. IV-05-029*, 26 July 2005.

(NRC, 2004a)
US Nuclear Regulatory Commission, *Protecting Our Nation Since 9-11-01, NUREG/BR-0314* (Washington, DC: Nuclear Regulatory Commission, September 2004).

(NRC, 2004b)
US Nuclear Regulatory Commission, "Commercial Irradiators", Fact Sheet, April 2004.

(NRC, 2003)
US Nuclear Regulatory Commission, "Dirty Bombs", Fact Sheet, March 2003.

(NRC, 1983)
US Nuclear Regulatory Commission, "Safety and Security of Irradiators", Information Notice No. 83-09, 9 March 1983.

(Nuclear News, 2005)
"Food Irradiation: License Sought for Hawaii Irradiator", *Nuclear News*, September 2005, pp 61-62.

(Ramberg, 1984)
Bennett Ramberg, *Nuclear Power Plants as Weapons for the Enemy: An Unrecognized Military Peril* (Berkeley, California: University of California Press, 1984).

(Raskin et al, 2002)
Paul Raskin et al, *Great Transition: The Promise and Lure of the Times Ahead* (Boston, Massachusetts: Stockholm Environment Institute, 2002).

(Reichmuth et al, 2005)
Barbara Reichmuth, Steve Short and Tom Wood, "Economic Consequences of a Rad/Nuc Attack: Cleanup Standards Significantly Affect Cost", a paper presented at, and recorded in a CD-ROM archive of, a conference sponsored by the US Department of Homeland Security, titled: "Working Together: R&D Partnerships in Homeland Security", 27-28 April 2005, Boston, Massachusetts.

(Stokstad, 2005)
Erik Stokstad, "Taking the Pulse of Earth's Life-Support Systems", *Science*, Volume 308, 1 April 2005, pp 41-43. (News story about the Millennium Ecosystem Assessment.)

(Thompson, 2004)
Gordon Thompson, testimony before the Public Utilities Commission of the State of California regarding Application No. 04-02-026, 13 December 2004. (This testimony, prepared for California Earth Corps, addressed the provision of an enhanced defense of Units 2 and 3 of the San Onofre Nuclear Generating Station.)

(Thompson, 2003)
Gordon Thompson, *Robust Storage of Spent Nuclear Fuel: A Neglected Issue of Homeland Security* (Cambridge, Massachusetts: Institute for Resource and Security Studies, January 2003).

(Walters, 2003)
William Walters, "An Overview of the Shaped Charge Concept", paper presented at the 11th Annual ARL/USMA Technical Symposium, 5 and 7 November 2003. (This symposium was sponsored by the Mathematical Sciences Center of Excellence at the US Military Academy (USMA) and hosted by the US Army Research Laboratory (ARL) and USMA.)

(Zimmerman and Loeb, 2004)
Peter D. Zimmerman and Cheryl Loeb, "Dirty Bombs: The Threat Revisited", *Defense Horizons* (a publication of the Center for Technology and National Security Policy, National Defense University), Number 38, January 2004, pp 1-11.

EXHIBIT PP

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ROBUST STORAGE OF SPENT NUCLEAR FUEL: A Neglected Issue of Homeland Security

by

Gordon Thompson

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A report commissioned by

Citizens Awareness Network

About IRSS

The Institute for Resource and Security Studies (IRSS) is an independent, non-profit corporation. It was founded in 1984 to conduct technical and policy analysis and public education, with the objective of promoting international security and sustainable use of natural resources. IRSS projects always reflect a concern for practical solutions to resource, environment and security problems. Projects include detailed technical studies, participation in public education and debate, and field programs that promote the constructive management of conflict.

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Dr. Gordon Thompson is the executive director of IRSS and a research professor at Clark University. He received an undergraduate education in science and mechanical engineering in Australia and a doctorate in applied mathematics from Oxford University. Thompson has extensive experience in assessing the safety and security hazards associated with nuclear facilities, and in identifying alternative designs and modes of operation that can reduce a facility's hazard potential.

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Abstract

The prevailing practice of storing most US spent nuclear fuel in high-density pools poses a very high risk. Knowledgeable attackers could induce a loss of water from a pool, causing a fire that would release to the atmosphere a huge amount of radioactive material. Nuclear reactors are also vulnerable to attack. Dry-storage modules used in independent spent fuel storage installations (ISFSIs) have safety advantages in comparison to pools and reactors, but are not designed to resist a determined attack. Thus, nuclear power plants and their spent fuel can be regarded as pre-deployed radiological weapons that await activation by an enemy. The US government and the Nuclear Regulatory Commission seem unaware of this threat.

This report sets forth a strategy for robust storage of US spent fuel. Such a strategy will be needed whether or not a repository is opened at Yucca Mountain. This strategy should be implemented as a major element of a defense-in-depth strategy for US civilian nuclear facilities. In turn, that defense-in-depth strategy should be a component of a homeland-security strategy that provides solid protection of our critical infrastructure.

The highest priority in a robust-storage strategy for spent fuel would be to re-equip spent-fuel pools with low-density, open-frame racks. As a further measure of risk reduction, ISFSIs would be re-designed to incorporate hardening and dispersal. Preliminary analysis suggests that a hardened, dispersed ISFSI could be designed to meet a two-tiered design-basis threat. The first tier would require high confidence that no more than a small release of radioactive material would occur in the event of a direct attack on the ISFSI by various non-nuclear instruments. The second tier would require reasonable confidence that no more than a specified release of radioactive material would occur in the event of attack using a 10-kilotonne nuclear weapon.

Table of Contents

1. Introduction

> "One fact dominates all homeland security threat assessments: terrorists are strategic actors. They choose their targets deliberately based on the weaknesses they observe in our defenses and our preparations. They can balance the difficulty in successfully executing a particular attack against the magnitude of loss it might cause."
> National Strategy for Homeland Security[1]

It is well known that nuclear power plants and their spent fuel contain massive quantities of radioactive material. (Note: Irradiated fuel discharged from a nuclear reactor is described as "spent" because it is no longer suitable for generating fission power.) Consequently, thoughout the history of the nuclear power industry, informed citizens have expressed concern that a substantial amount of this material could be released to the environment. One focus of concern has been the possibility of an accidental release caused by human error, equipment failure or natural forces (e.g., an earthquake). In response to citizens' demands and events such as the Three Mile Island reactor accident of 1979, the US Nuclear Regulatory Commission (NRC) has taken some actions that address this threat.

To date, citizens have been much less successful in forcing the NRC to address a related threat -- the possibility that a release of radioactive material will be caused by an act of malice or insanity. The citizens' failure is not for lack of effort. For many years, citizen groups have petitioned the NRC and engaged in licensing interventions, seeking to persuade the NRC to address this threat. Yet, the agency has responded slowly, reluctantly and in limited ways, even after the terrorist attacks of 11 September 2001. This limited response is not unique to the NRC. The US government in general seems unwilling to address the possibility that an enemy, domestic or foreign, will exploit a civilian nuclear facility as a radiological weapon.

The terrorist attacks of September 2001 demonstrated the vulnerability of our industrial society to determined acts of malice, and cruelly validated long-neglected warnings by many analysts and concerned citizens. In response, the United States employed its military capabilities in Afghanistan and has signaled its willingness to use those capabilities in Iraq and elsewhere. Yet, nothing significant has been done to defend US nuclear power plants and their spent fuel against attack. There is much discussion in the media about "dirty bombs" that disperse radioactive material, but decision makers seem

[1] Office of Homeland Security, 2002, page 7.

largely unaware that civilian nuclear facilities contain massive quantities of radioactive material and are vulnerable to attack.

What is Robust Storage?

This report addresses robust storage of spent fuel from nuclear power plants. Here, the term "robust" means that a facility for storing spent fuel is made resistant to attack. The provision of robust storage would substantially reduce the potential for a maliciously-induced release of radioactive material from spent fuel, and would thereby enhance US homeland security. Robust storage of spent fuel should be viewed as a component of a national strategy for reducing the vulnerability of all civilian nuclear facilities, within the context of homeland security. This report takes such a view.

A spent-fuel-storage facility can be made resistant to attack in three ways. First, the facility can be made passively safe, so that spent fuel remains in a safe state without needing electrical power, cooling water or the presence of an operating crew. Second, the facility can be "hardened", so that the spent fuel and its containment structure are protected from damage by an instrument of attack (e.g., an anti-tank missile). For a facility at ground level, hardening involves the provision of layers of concrete, steel, gravel or other materials above and around the spent fuel. Third, the facility can be "dispersed", so that spent fuel is not concentrated at one location, but is spread more uniformly across the site. Dispersal can reduce the magnitude of the radioactive release that would arise from a given attack.

At present, all but a tiny fraction of US spent fuel is stored at the nation's nuclear power plants. Most of this fuel is stored at high density in water-filled pools that are adjacent to, but outside, the containments of the reactors. This mode of storage does not meet any of the above-stated three conditions for robustness. High-density spent-fuel pools are not passively safe. Indeed, if water is lost from such a pool, which could occur in various ways, the fuel will heat up, self-ignite and burn, releasing a large amount of radioactive material to the environment. Spent-fuel pools are not hardened against attack, and a pool concentrates a large amount of spent fuel in a small space, which is the antithesis of dispersal.

A growing fraction of US spent fuel, now about 6 percent of the total inventory, is stored in dry-storage facilities at nuclear power plants. The storage is "dry" in the sense that the spent fuel is surrounded by a gas such as helium, rather than by water. The NRC describes a spent-fuel-storage facility, other than a spent-fuel pool at a nuclear power plant, as an independent spent fuel storage installation (ISFSI).[2] All but two of the existing ISFSIs are at the sites of nuclear power plants, either operational plants or plants undergoing decommissioning.[3] Future ISFSIs could be built at nuclear-power-plant sites or at away-from-reactor sites. An application to build an ISFSI at an away-from-reactor site -- Skull Valley, Utah -- is awaiting decision by the NRC. It should be noted that the nuclear industry is building dry-storage ISFSIs not as an alternative to high-density pools, but to accommodate the growing inventory of spent fuel as pools become full.

Dry-storage ISFSIs meet one of the above-stated three conditions for robust storage of spent fuel. They are passively safe, because their cooling depends on the natural circulation of ambient air. However, none of the existing or proposed ISFSIs is hardened, and none of them is dispersed across its site.

A Broader Context

This report describes the need for robust storage of all US spent fuel, whether in pools or dry-storage ISFSIs, and sets forth a strategy for meeting this need. As discussed above, a productive discussion of these issues must occur within a broader context, which is is addressed in this report. The provision of robust storage of spent fuel must be viewed as a component of a national strategy for defending the nation's civilian nuclear industry, including all of the nuclear power plants and all of their spent fuel. That strategy must in turn be viewed as a component of homeland security in general. Finally, homeland security must be viewed as a key component of US strategy for national defense and international security.

The various levels of security, ranging from the security of nuclear facilities to the security of the nation and the international community, are linked in surprising ways. If our nuclear facilities and other parts of our infrastructure -- such as the airlines -- are poorly defended, we may feel compelled to use military force aggressively around the world, to punish or pre-empt attackers. Such action poses the risk of arousing hostility and promoting anarchy, leading to new attacks on our homeland. The potential exists for an escalating spiral of violence. If, however, our nuclear facilities and other critical items of infrastructure are strongly defended, we can gain a double benefit. First, the communities around each facility will receive direct protection. Second, we can take a more measured approach to national defense, with a greater prospect of detecting, deterring and apprehending potential attackers without undermining civil liberties or international

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[2] One wet-storage ISFSI exists in the USA, at Morris, Illinois. All other existing ISFSIs, and all planned ISFSIs, employ dry storage.

[3] The existing ISFSIs that are not at nuclear-power-plant sites are the small wet-storage facility at Morris and a facility in Idaho that stores fuel debris from Three Mile Island Unit 2.

security. Thus, a decision about the level of protection to be provided at a nuclear facility has wide-ranging implications.

The Need for Further Investigation

The investigation leading to this report has identified a number of technical issues that could not be resolved within the scope of the investigation. Issues of this kind are flagged in relevant parts of the report. Also, this report has a broad focus. It sets forth a strategy for providing robust storage of US spent fuel, and outlines a design approach for hardened, dispersed, dry storage. Additional analysis, supported by experiments, would be needed to test and refine this design approach and to determine the feasibility of implementing hardened, dispersed, dry storage at particular sites. That work would, in turn, set the stage for detailed, engineering-design studies that could lead to site-specific implementation. Moreover, a variety of governmental actions would be needed to support nationwide implementation of robust storage. For example, the NRC would need to develop new regulations and guidance. Also, the implementation program would require new financing arrangements, which would probably require new legislation.

Sensitive Information

An attack on a nuclear facility could be assisted by detailed information about the facility's vulnerability and the measures taken to defend the facility. Thus, certain categories of information related to a facility are not appropriate for general distribution. However, experience shows that secrecy breeds incompetence, complacency and conflicts of interest within the organizations that are shielded from public view.[4] Thus, in the context of defending nuclear facilities, protection of the public interest requires that secrecy be limited in two respects. Firstly, the only information that should be withheld from the public is detailed technical information that would directly assist an attacker. Second, stakeholder groups should be fully engaged in the development and implementation of measures for defending nuclear facilities, through processes that allow debate but protect sensitive information.[5] It should be noted that this report does not contain sensitive information and is suitable for general distribution.

[4] Thompson, 2002a, Section X.
[5] Thompson, 2002a, Sections IX and X.

Robust Storage and Related Concepts

Issues addressed in this report have been the subject of public debate around the United States, and this debate has been framed in a number of ways. One approach has been to speak of "risk reduction", whereby robust storage of spent fuel and related measures are used to reduce the risk of a maliciously-induced release of radioactive material from nuclear facilities. This approach explicitly recognizes that the risk can be reduced but, given the continued existence of radioactive material, cannot be eliminated. Another approach has been to speak of "hardened on-site storage" as a strategy for managing US spent fuel. This approach advocates the robust storage of all spent fuel, but only at the sites of nuclear power plants. A related but distinct approach is "nuclear guardianship", whose supporters argue that radioactive materials should be contained in accessible, monitored storage facilities for the foreseeable future. The robust-storage strategy that is outlined in this report is compatible with all three approaches, and with a prudent assessment of the likelihood and timeframe for development of a radioactive-waste repository at Yucca Mountain.

Structure of this Report

The remainder of this report begins, in Section 2, with the provision of some basic information about US nuclear power plants and their spent fuel. Then, Section 3 discusses the potential for attacks on nuclear facilities, describes the US government's response to this threat, and outlines a balanced response. Section 4 addresses the vulnerability of nuclear facilities to attack, describes the potential consequences of an attack, outlines a defense-in-depth strategy for a nuclear facility, and sets forth a national strategy for robust storage of spent fuel. Elaborating upon this proposed strategy for robust storage, Section 5 discusses the various factors that must be considered in planning hardened, dispersed, dry storage of spent fuel. Section 6 offers a design approach that accounts for these factors. A set of requirements for nationwide implementation of robust storage is described in Section 7. Conclusions are set forth in Section 8, and a bibliography is provided in Section 9. Documents cited in this report are, unless indicated otherwise, drawn from this bibliography.

2. Nuclear Power Plants and Spent Fuel in the USA

2.1 Status and Trends of Nuclear Power Plants and Spent Fuel

There are 103 commercial nuclear reactors operating in the USA at 65 sites in 31 states.[6] Of these 103 reactors, 69 are pressurized-water reactors (PWRs), 9 with ice-condenser containments and 60 with dry containments. The remaining 34 reactors are boiling-water reactors (BWRs), 22 with Mark I containments, 8 with Mark II containments and 4 with Mark III containments. In addition there are 27 previously-operating commercial reactors in various stages of storage or decommissioning. As of December 2000, all but 2 of the 103 operating reactors had been in service for at least 9 years, and 55 reactors had been in service for at least 19 years.[7] Thus, the reactor fleet is aging. The nominal duration of a reactor operating license is 40 years.

Four of the 103 operating reactors have design features intended to resist aircraft impact. The Limerick Unit 1, Limerick Unit 2 and Seabrook reactors were designed to withstand the impact of an aircraft weighing 6 tonnes, while the Three Mile Island Unit 1 reactor was designed to withstand the impact of an aircraft weighing 90 tonnes. No other US reactor was designed to withstand aircraft impact.[8]

Wet and Dry Storage of Spent Fuel

The core of a commercial nuclear reactor consists of several hundred fuel assemblies.[9] Each fuel assembly contains thousands of cylindrical, uranium-oxide pellets stacked inside long, thin-walled tubes made of zirconium alloy. These tubes are often described as the "cladding" of the fuel. After several years of use inside an operating reactor, a fuel assembly becomes "spent" in the sense that it is no longer suitable for generating fission power. Then, the fuel is discharged from the reactor and placed in a water-filled pool adjacent to the reactor but outside the reactor containment. This fuel, although spent, contains numerous radioactive isotopes whose decay generates ionizing radiation and heat.

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[6] In addition, Browns Ferry Unit 1, a BWR with a Mark I containment, is nominally operational. However, it is defueled and not in service.
[7] Data from the NRC website (www.nrc.gov), 24 April 2002.
[8] Markey, 2002, page 73.
[9] The number of fuel assemblies in a reactor core ranges from 121 (in some PWRs) to 764 (in some BWRs).

After a period of storage in a pool, the thermal power produced by a fuel assembly declines to a level such that the assembly can be transferred to a dry-storage ISFSI. Current practice is to allow a minimum cooling period of 5 years before transfer to dry storage. However, this cooling period reflects an economic and safety tradeoff rather than a fundamental physical limit. Fuel cooled for a shorter period than 5 years could be transferred to dry storage, but in that case fewer assemblies could be placed in each dry-storage container. Alternatively, older and younger spent fuel (counting age from the date of discharge from the reactor) could be co-located in a dry-storage container. The major physical limit to placement of spent fuel in dry storage is the maximum temperature of the cladding, which the NRC now sets at 400 degrees C. This temperature limit constrains the allowable heat output of the fuel, which in turn constrains the cooling period.

Development of ISFSIs

At present, there are 20 ISFSIs in the USA, of which 15 are at sites where commercial reactors are in operation.[10] More ISFSIs will be needed, because the spent-fuel pools at operating reactors are filling up. Analysis by Allison Macfarlane of MIT shows that, by 2005, almost two-thirds of reactor licensees will face the need to acquire onsite dry-storage capacity, even if shipment of spent fuel away from the reactor sites begins in 2005.[11] NAC International, a consulting firm and vendor of dry-storage technology, reaches similar conclusions. NAC estimates that, at the end of 2000, about 6 percent of the US inventory of commercial spent fuel was stored in ISFSIs at reactor sites, whereas about 30 percent of the inventory will be stored in ISFSIs by 2010.[12] New ISFSIs entering operation by 2010 will generally be at reactor sites, although some might be at new sites. At present, only one proposed ISFSI at a new site -- Skull Valley, Utah -- seems to be a plausible candidate for operation by 2010.

Shipment of Spent Fuel from Reactor Sites

If spent fuel is shipped away from a reactor site, the fuel could have three possible destinations. First, fuel could be shipped to another reactor site, which Carolina Power and Light Co. is now doing, shipping fuel from its

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[10] Data from the NRC website (www.nrc.gov), 24 April 2002.
[11] Macfarlane, 2001a.
[12] NAC, 2001. NAC estimates that the end-2000 US inventory of spent fuel was 42,900 tonnes, of which 2,430 tonnes was in ISFSIs. Also, NAC estimates that the 2010 US inventory will be 64,300 tonnes, of which 19,450 tonnes will be in ISFSIs.

Brunswick and Robinson reactors to its Harris site.[13] Second, fuel could be shipped to an ISFSI at an away-from-reactor site, such as Skull Valley. Third, fuel could be shipped to a repository at Yucca Mountain, Nevada. At Yucca Mountain, the fuel would be emplaced in underground tunnels. Under some scenarios for the operation of Yucca Mountain, emplacement would be preceded by a period of interim storage at the surface.

There seems to be no current planning for shipment of spent fuel to any reactor site other than Harris. Also, there are factors that argue against shipping fuel to an away-from-reactor ISFSI. First, such shipment would increase the overall transport risk, because fuel would be shipped twice, first from the reactor site to the ISFSI, and then from the ISFSI to the ultimate repository. Second, an away-from-reactor ISFSI would hold a comparatively large inventory of spent fuel, creating a potentially attractive target for an enemy.[14] Third, shipment to an away-from-reactor ISFSI would not free most reactor licensees from the obligation to build some ISFSI capacity at each reactor site.[15] Fourth, there is a risk that a large, away-from-reactor ISFSI would become, by default, a permanent repository, despite having no long-term containment capability. Finally, storage of spent fuel in reactor-site ISFSIs could be cheaper than shipping fuel to away-from-reactor ISFSIs.[16] Time will reveal the extent to which these factors affect the development of away-from-reactor ISFSIs at Skull Valley or elsewhere.

Yucca Mountain

The Yucca Mountain repository project will not free reactor licensees from the obligation to develop ISFSI capacity, for three reasons. First, the Yucca Mountain repository may never open. This project is politically driven, does not have a sound scientific basis, and is going forward only because previously-specified technical criteria for a repository have been abandoned.[17] These deficiencies add weight to the determined opposition to this project by the state of Nevada and other entities. That opposition will also be fueled by concern about the risk of transporting fuel to Yucca Mountain. Second, decades will pass before fuel can be emplaced in a repository at Yucca Mountain. The US Department of Energy (DOE) claims that it can open the repository in 2010, but the US General Accounting Office has determined that

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[13] The Harris site features one reactor and four spent-fuel pools, and thus has more pool-storage capacity than other reactor sites. Spent fuel that is shipped to Harris is placed in a pool, and there is no current plan to build an ISFSI at Harris.

[14] The proposed Skull Valley ISFSI could hold 40,000 tonnes of spent fuel, according to the Private Fuel Storage website (www.privatefuelstorage.com), 4 October 2002.

[15] Macfarlane, 2001a.

[16] Macfarlane, 2001b.

[17] Ewing and Macfarlane, 2002.

several factors, including budget limitations, could extend this date to 2015 or later.[18] DOE envisions that, after the repository is opened, emplacement of fuel will occur over a period of at least 24 years and potentially 50 years.[19] This vision may prove to be optimistic. Third, under present federal law the Yucca Mountain repository will hold no more than 63,000 tonnes of commercial spent fuel.[20] Yet, the cumulative amount of commercial spent fuel to be generated during the lifetimes of the 103 currently-licensed reactors is likely to exceed 80,000 tonnes.[21] Reactor licensees have shown strong interest in obtaining license extensions which, if granted, would lead to the production of a substantial additional amount of spent fuel.

Summary

To summarize the preceding paragraphs, it is clear that thousands of tonnes of spent fuel will be stored at reactor sites for several decades to come, in pools and/or ISFSIs. Similar amounts of fuel might be stored at away-from-reactor ISFSIs. Moreover, it is entirely possible that the Yucca Mountain repository will not open, with the result that the entire national inventory of spent fuel will be stored for decades, perhaps for 100 years or more, at reactor sites (in pools and/or ISFSIs) and/or at away-from-reactor ISFSIs. It is therefore imperative that each ISFSI is planned to allow for its possible extended use. The NRC has begun to recognize this need, by performing research to determine if dry storage of spent fuel can safely continue for a period of up to 100 years.[22]

2.2 Present Practice for Storing Spent Fuel

The technology that is currently used for storing spent fuel was developed without consideration of the possibility of an attack. Nor was there any consideration of the possibility that spent fuel would be stored for many decades. Instead, the technology has developed incrementally, in response to

[18] Jones, 2002b.

[19] DOE, 2002. DOE contemplates the construction of a surface facility for interim storage of spent fuel at Yucca Mountain, especially if emplacement of fuel occurs over a period of 50 years. However, given the cost of this surface facility, a more likely alternative is that fuel would remain in ISFSIs until it could be emplaced in the repository.

[20] DOE, 2002. The Nuclear Waste Policy Act limits the total amount of waste that can be placed in a first repository to 70,000 tonnes until a second repository is in operation. DOE plans to use 63,000 tonnes of this capacity for commercial spent fuel. DOE has studied the possible expansion of Yucca Mountain's capacity to include 105,000 tonnes of commercial spent fuel together with other wastes.

[21] Macfarlane, 2001a.

[22] "Radioactive Waste Safety Research", from NRC website (www.nrc.gov), 23 September 2002.

changing circumstances. Throughout this process, cost minimization has been a top priority.

When the present generation of nuclear power plants was designed, the nuclear industry and the US government both assumed that spent nuclear fuel would be reprocessed. Thus, spent-fuel pools were designed to hold only the amount of spent fuel that a reactor would discharge over a period of a few years. This was accomplished by equipping the pools with low-density, open-frame racks. However, in the mid-1970s the US government banned reprocessing, and the industry faced the prospect of an accumulating inventory of spent fuel.

High-Density Spent-Fuel Pools

Industry's response to growing spent-fuel inventories has been to re-rack spent-fuel pools at progressively higher densities, so that more fuel can be stored in a given pool. Now, pools across the nation are equipped with high-density, closed-frame racks that, in many instances, fill the floor area of the pool from wall to wall. The NRC has allowed this transition to occur despite the fact that a loss of water from a pool equipped with high-density racks can cause the zirconium cladding of the spent fuel to heat up, spontaneously ignite and burn, releasing a large amount of radioactive material to the atmosphere. This hazard is discussed further in Section 4.2.

Dry Storage as a Supplement to High-Density Pools

Consistent with the focus on cost minimization, the nuclear industry has turned to alternative methods of fuel storage only when pools have begun to fill up. Preventing a pool fire has not been a consideration. Thus, dry-storage ISFSIs have not been introduced as an alternative to high-density pool storage. Instead, standard industry practice is to fill a pool to nearly its maximum capacity, then to transfer older spent fuel from the pool to an ISFSI at a rate just sufficient to open up space in the pool for fuel that is discharged from the reactor.[23]

As a part of this strategy, each ISFSI has a modular design. One or more concrete pads are laid in the open air. Each pad supports an array of identical fuel-storage modules that are purchased and installed as needed, so that the ISFSI grows incrementally. Additional pads can be laid as needed.

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[23] In standard practice, the maximum storage capacity of a spent-fuel pool is less than the number of fuel-assembly slots in the pool, to allow for the possibility of offloading a full reactor core. However, preserving the capacity for a full-core offload is not a licensing requirement.

This modular approach to the development of ISFSIs has functional and cost advantages. However, the present implementation of the approach is not driven by security considerations, and is therefore proceeding slowly. Pools remain packed with fuel at high density, and can therefore be readily exploited as radiological weapons. Moreover, the ISFSIs themselves are not designed to resist attack.

Types of Dry-Storage Module

The NRC has approved 14 different designs of dry-storage module for general use in ISFSIs.[24] In each of these designs, the central component of the module is a cylindrical, metal container whose interior is equipped with a metal basket structure into which spent fuel assemblies can be inserted. This container is filled with spent fuel while immersed in a spent-fuel pool. Then, the container's lid is attached, the container is removed from the pool and sealed, its interior is dried and filled with an inert gas (typically helium), and it is transferred to the ISFSI.

Available designs of dry-storage modules for ISFSIs fall into two basic categories. In the first category, the metal container has a thick wall, and no enclosing structure is provided. This type of module is commonly described as a "monolithic cask". In the second category, the metal container has a thin wall and is surrounded by an overpack. Different overpacks are used during the three phases of spent-fuel management. First, during the initial transfer of fuel from a spent-fuel pool to an onsite ISFSI, the metal container is surrounded by a transfer overpack. Second, during storage in an ISFSI, the metal container is surrounded by a storage overpack. Third, if fuel is eventually shipped away from the site, the metal container would be placed inside a transport overpack. The second category of module is described here as an "overpack system".

A Typical Monolithic Cask

One example of a monolithic cask is the CASTOR V/21, which was approved by the NRC in 1990 for general use and is employed at the Surry ISFSI. This cask is about 4.9 meters long and 2.4 meters in diameter, and can hold 21 PWR fuel assemblies. In the storage position the cask axis is vertical. The cask body is made of ductile cast iron with a wall thickness of about 38 cm. Circumferential fins on the outside of the cask body facilitate cooling by natural circulation of ambient air. Fully loaded, this cask weighs about 98 tonnes.[25] The NRC has approved this cask for storage but not for transport,

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[24] "Dry Spent Fuel Storage Designs: NRC Approved for General Use", from NRC website (www.nrc.gov), 20 September 2002.
[25] Raddatz and Waters, 1996.

although CASTOR casks are widely used in Europe for both purposes. CASTOR casks have not been popular in the US market.

Examples of Overpack Systems

One example of an overpack system is the NUHOMS design, which the NRC approved for general use in 1995. In this design, the metal container that holds the spent fuel is about 4.7 meters long and 1.7 meters in diameter, and has a wall thickness of 1.6 cm. This container, which is placed horizontally inside its storage overpack, is made of stainless steel and can hold 24 PWR fuel assemblies or 52 BWR fuel assemblies. The storage overpack is a reinforced-concrete box about 6.1 meters long, 4.6 meters high and 2.7 meters wide, with walls and roof 91 cm thick.[26] Ambient air passes into and out of this structure through vents, and cools the metal container by natural convection. NUHOMS modules are in use at the Davis-Besse site and some other reactor sites.

A second example of an overpack system is the NAC-UMS, which the NRC approved for general use in 2000. In this instance, the metal container is about 4.7 meters long and 1.7 meters in diameter, and has a wall thickness of 1.6 cm. This container, which is made of stainless steel, can hold 24 PWR fuel assemblies or 56 BWR fuel assemblies. The storage overpack is a vertical-axis reinforced-concrete cylinder about 5.5 meters high and 3.5 meters in diameter. The wall of this overpack consists of a steel liner 6.4 cm thick and a layer of concrete 72 cm thick. Ambient air passes into and out of the overpack through vents, and cools the metal container by natural convection. At the Maine Yankee nuclear power plant, which is being decommissioned, sixty NAC-UMS modules are being installed. Most of the modules will be used to store spent fuel discharged from the plant. Some modules will store pieces of the reactor core shroud, which is classified as greater-than-Class C (GTCC) waste.[27]

Monolithic Casks versus Overpack Systems

The two categories of dry-storage module employ distinct design approaches. In a monolithic cask such as the CASTOR, spent fuel is contained within a thick-walled metal cylinder that is comparatively robust.[28] In an overpack system the fuel is contained within a thin-walled metal container that has a

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[26] Ibid.

[27] Stone and Webster, 1999.

[28] The vendor of the CASTOR cask has developed a cheaper type of monolithic cask that is made as a steel-concrete-steel sandwich. This cask, known as CONSTOR, was developed for storage and transport of spent fuel from Russian reactors. The vendor states that the CONSTOR cask could be used in the USA. See: Peters et al, 1999.

limited capability to withstand impact, fire or corrosion. The storage overpack employs concrete -- a cheap material -- as its primary constituent. The transfer and transport overpacks can be used multiple times. Thus, an overpack system can be substantially cheaper -- about half as expensive per fuel assembly, according to some reports -- than a monolithic cask.

ISFSI Configuration

At ISFSIs in the USA, dry-storage modules are placed on concrete pads in the open air. This approach contrasts with German practice, where dry-storage modules -- usually CASTOR casks -- are placed inside buildings. These buildings are designed to have some resistance to attack from outside using anti-tank weapons. This aspect of their design has been informed by tests conducted in the period 1979-1980. At one German reactor site -- Neckarwestheim -- the ISFSI is inside a tunnel built into the side of a hill.[29]

Another feature of the US approach to ISFSI design, consistent with the high priority assigned to cost minimization, is that dry-storage modules are packed closely together in large numbers. In illustration, consider the ISFSI that is proposed for the Diablo Canyon site in California. This facility would hold up to 140 of Holtec's HI-STORM 100 dry-storage modules, whose design is similar to the NAC-UMS system described above. These modules would sit on concrete pads, 20 casks per pad in a 4 by 5 array. Initially, two pads would be built. Ultimately, as the ISFSI expanded, seven pads would be positioned side by side, covering an area about 150 meters by 32 meters. Each module would be a vertical-axis cylinder about 3.7 meters in diameter and 5.9 meters high. The center-to-center spacing of modules would be about 5.5 meters, leaving a gap of 1.8 meters between modules. A security fence would surround the area needed for this array, at a distance of about 15 meters from the outermost modules. That fence would in turn be surrounded by a second fence, at a distance of about 30 meters from the outermost modules.[30]

2.3 Present Security Arrangements

One could reasonably expect that the defense strategy for a nuclear-facility site would be a component of a strategy for homeland security, which would itself be a component of an overall strategy for national security. Moreover, one could expect that the site-level strategy would provide a defense in depth. (See Section 4.4 of this report for an explanation of defense in depth.) Logical planning of this kind may eventually occur. However, at present, the security

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[29] Janberg, 2002.
[30] PG&E, 2001a.

arrangements for US nuclear facilities are not informed by any strategic vision.

Differing Positions on the Threat of Attack

For several decades it has been clear to many people that nuclear power plants and other commercial nuclear facilities are potential targets of acts of malice or insanity, including highly destructive acts. The NRC has repeatedly rebuffed citizens' requests that this threat be given the depth of analysis that would be expected, for example, in an environmental impact statement (EIS).[31] This history is illustrated by a September 1982 ruling by the Atomic Safety and Licensing Board (ASLB) in the operating-license proceeding for the Harris plant. The intervenor, Wells Eddleman, had proffered a contention alleging, in part, that the plant's safety analysis was deficient because it did not consider the "consequences of terrorists commandeering a very large airplane.....and diving it into the containment." In rejecting this contention the ASLB stated:[32]

> "This part of the contention is barred by 10 CFR 50.13. This rule must be read *in pari materia* with 10 CFR 73.1(a)(1), which describes the "design basis threat" against which commercial power reactors *are* required to be protected. Under that provision, a plant's security plan must be designed to cope with a violent external assault by "several persons," equipped with light, portable weapons, such as hand-held automatic weapons, explosives, incapacitating agents, and the like. Read in the light of section 73.1, the principal thrust of section 50.13 is that military style attacks with heavier weapons are not a part of the design basis threat for commercial reactors. Reactors could not be effectively protected against such attacks without turning them into virtually impregnable fortresses at much higher cost. Thus Applicants are not required to design against such things as artillery bombardments, missiles with nuclear warheads, or kamikaze dives by large airplanes, despite the fact that such attacks would damage and may well destroy a commercial reactor."

In this statement, the ASLB correctly described the design basis for US nuclear power plants. However, other design bases are possible. In the early 1980s the

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31 In illustration of this continuing policy, on 18 December 2002 the NRC Commissioners dismissed four licensing interventions calling for EISs that consider the potential for malicious acts at nuclear facilities. One intervention, by the state of Utah, addressed the proposed ISFSI at Skull Valley. The other three interventions, by citizen groups, addressed: a proposed spent-fuel-pool expansion at Millstone Unit 3; a proposed MOX-fuel-fabrication facility; and proposed license renewals for the McGuire and Catawba nuclear power plants.

32 ASLB, 1982.

reactor vendor ASEA-Atom developed a preliminary design for a commercial reactor known as the PIUS reactor. The design basis for the PIUS reactor included events such as equipment failures, operator errors and earthquakes, but also included: (i) takeover of the plant for one operating shift by knowledgeable saboteurs equipped with large amounts of explosives; (ii) aerial bombardment with 1,000-pound bombs; and (iii) abandonment of the plant by the operators for one week.[33] It seems likely that this design basis would also provide protection against a range of other assaults, including the impact of a large, fuel-laden aircraft. Clearly, ASEA-Atom foresaw a world in which acts of malice could pose a significant threat to nuclear facilities. The NRC has never exercised an equivalent degree of foresight.

A Brief History

Some US nuclear facilities have been specifically designed to resist attack. For example, in the early 1950s five heavy-water reactors were built at the Savannah River site in South Carolina, to produce plutonium and tritium for use in US nuclear weapons. In order to resist an attack by the USSR using nuclear weapons, the reactors were dispersed across a large site and hardened against blast. The reactor buildings were designed to withstand an external blast of 7 psi, the overpressure that could be experienced at about 2 miles from a 1-megatonne surface burst. However, the purpose was to preserve the reactors' ability to produce weapons material after an attack, rather than to protect the public from a release of radioactive material. Indeed, these reactors had minimal safety systems when they first entered service. Safety systems were added over the years, but the reactors' safety standards never approached the level that is expected for commercial reactors.[34]

In 1950, the Reactor Safeguards Committee of the US Atomic Energy Commission (AEC) produced a report -- designated WASH-3 -- that considered the potential for reactor accidents and estimated the offsite effects of an accident. This report gave special attention to sabotage as a potentially important cause of reactor accidents. About 16 years later, during the construction license proceedings for Turkey Point Units 3 and 4 in Florida, an intervenor raised the question of an attack on these nuclear power plants from a hostile country (i.e., Cuba). The AEC held that it was not responsible for providing protection against such an attack.[35] This position remains enshrined in the NRC's regulation 10 CFR 50.13, which states:[36]

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[33] Hannerz, 1983.
[34] Thompson and Sholly, 1991.
[35] Okrent, 1981, pp 18-19.
[36] NRC Staff, 2002.

> "An applicant for a license to construct and operate a production or utilization facility, or for an amendment to such license, is not required to provide for design features or other measures for the specific purpose of protection against the effects of (a) attacks and destructive acts, including sabotage, directed against the facility by an enemy of the United States, whether a foreign government or other person, or (b) use or deployment of weapons incident to US defense activities."

Pursuant to this regulation, the NRC's licensees are not required to design or operate nuclear facilities to resist enemy attack. However, events have forced the NRC to progressively modify this position, so as to require greater protection against acts of malice or insanity. A series of incidents, including the 1993 bombing of the World Trade Center in New York, eventually forced the NRC to introduce, in 1994, regulations requiring licensees to defend nuclear power plants against vehicle bombs. The terrorist events of 11 September 2001 forced the NRC to require additional, interim measures by licensees to protect nuclear facilities, and are also forcing the NRC to consider strengthening its regulations in this area. Nevertheless, present NRC regulations require only a light defense of nuclear facilities.

NRC Regulations for Defending Nuclear Facilities

Present NRC regulations for the defense of nuclear facilities are focused on site security. As described in Section 4.4, below, site security is one of four types of measure that, taken together, could provide a defense in depth against acts of malice or insanity. The other three types of measure are, with some limited exceptions, ignored in present NRC regulations and requirements.[37]

At a nuclear power plant or an ISFSI, the NRC requires the licensee to implement a set of physical protection measures. According to the NRC, these measures provide defense in depth by taking effect within defined areas with increasing levels of security. In fact, these measures provide only a fraction of the protection that could be provided by a comprehensive defense-in-depth strategy. Within the outermost physical protection area, known as the Exclusion Area, the licensee is expected to control the area but is not required to employ fences and guard posts for this purpose. Within the Exclusion area is a Protected Area encompassed by physical barriers including one or more fences, together with gates and barriers at points of entry. Authorization for unescorted access within the Protected Area is based on background and behavioral checks. Within the Protected Area are Vital

[37] For information about the NRC's present regulations and requirements for nuclear-facility defense, see: the NRC website (www.nrc.gov) under the heading "Nuclear Security and Safeguards", 2 September 2002; Markey, 2002; Meserve, 2002; and NRC, 2002.

Areas and Material Access Areas that are protected by additional barriers and alarms; unescorted access to these locations requires additional authorization.

Associated with the physical protection areas are measures for detection and assessment of an intrusion, and for armed response to an intrusion. Measures for intrusion detection include guards and instruments whose role is to detect a potential intrusion and notify the site security force. Then, security personnel seek additional information through means such as direct observation and closed-circuit TV cameras, to assess the nature of the intrusion. If judged appropriate, an armed response to the intrusion is then mounted by the site security force, potentially backed up by local law enforcement agencies and the FBI.

The Design Basis Threat

The design of physical protection areas and their associated barriers, together with the design of measures for intrusion detection, intrusion assessment and armed response, is required to accommodate a "design basis threat" (DBT) that is specified by the NRC in 10 CFR 73.1. The DBT for an ISFSI is less demanding than that for a nuclear power plant. At a nuclear power plant, the dominant sources of hazard are the reactor and the spent-fuel pool(s). In theory, both of these items receive the same level of protection, but in practice the reactor has been the main focus of attention. At present, the DBT for a nuclear power plant has the following features:[38]

> "(i) A determined violent external assault, attack by stealth, or deceptive actions, of several persons with the following attributes, assistance and equipment: (A) Well-trained (including military training and skills) and dedicated individuals, (B) inside assistance which may include a knowledgeable individual who attempts to participate in a passive role (e.g., provide information), an active role (e.g., facilitate entrance and exit, disable alarms and communications, participate in violent attack), or both, (C) suitable weapons, up to and including hand-held automatic weapons, equipped with silencers and having effective long range accuracy, (D) hand-carried equipment, including incapacitating agents and explosives for use as tools of entry or for otherwise destroying reactor, facility, transporter, or container integrity or features of the safeguards system, and (E) a four-wheel drive land vehicle used for transporting personnel and their hand-carried equipment to the proximity of vital areas, and

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[38] 10 CFR 73.1, Purpose and Scope, from the NRC web site (www.nrc.gov), 2 September 2002.

(ii) An internal threat of an insider, including an employee (in any position), and

(iii) A four-wheel drive land vehicle bomb."

For an ISFSI, the DBT is the same as for a nuclear power plant except that it does not include the use of a four-wheel-drive land vehicle, either for transport of personnel and equipment or for use as a vehicle bomb. This is true whether the ISFSI is at a new site or a reactor site. Thus, an ISFSI at a reactor site will be less protected than the reactor(s) and spent-fuel pool(s) at that site. At a reactor site or a new site, an ISFSI will be vulnerable to attack by a vehicle bomb. (Note: An NRC order of October 2002 to reactor-site ISFSI licensees, as discussed below, might require vehicle-bomb protection at reactor-site ISFSIs. Measures required by this order have not been disclosed.)

Interim, Additional Requirements by the NRC

After the events of 11 September 2001, the NRC concluded that its requirements for nuclear power plant security were inadequate. Accordingly, the NRC issued an order to licensees of operating plants in February 2002, and similar orders to licensees of decommissioning plants in May 2002 and reactor-site ISFSI licensees in October 2002, requiring "certain compensatory measures", also described as "prudent, interim measures", whose purpose is to "provide the Commission with reasonable assurance that the public health and safety and common defense and security continue to be adequately protected in the current generalized high-level threat environment".[39] The additional measures required by these orders have not been publicly disclosed, but the NRC Chairman has stated that they include:[40]

(i) increased patrols;
(ii) augmented security forces and capabilities;
(iii) additional security posts;
(iv) vehicle checks at greater stand-off distances;
(v) enhanced coordination with law enforcement and military authorities;
(vi) additional restrictions on unescorted access authorizations;
(vii) plans to respond to plant damage from explosions or fires; and
(viii) assured presence of Emergency Plan staff and resources.

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[39] The quoted language is from page 2 of the NRC's order of 25 February 2002 to all operating power reactor licensees. Almost-identical language appears in the NRC's orders of 23 May 2002 to all decommissioning power reactor licensees and 16 October 2002 to all ISFSI licensees who also hold 10 CFR 50 licenses.

[40] Meserve, 2002.

In addition to requiring these additional security measures, the NRC has established a Threat Advisory System that warns of a possible attack on a nuclear facility. This system uses five color-coded threat conditions ranging from green (low risk of attack) to red (severe risk of attack). These threat conditions conform with those used by the Office of Homeland Security. Also, the NRC is undertaking what it describes as a "top-to-bottom review" of its security requirements. The NRC has stated that it expects that this review will lead to revision of the present DBT. The review is not proceeding on any specific schedule.

Limitations of the Design Basis Threat

A cursory examination of the present DBT reveals significant limitations. For example, this threat does not include aircraft bombs (e.g., fuel-laden commercial aircraft, light aircraft packed with high explosive) or boat bombs.[41] This threat does not include lethal chemical weapons as instruments for disabling security personnel. This threat allows for one vehicle bomb, but not for a subsequent vehicle bomb that gains access to a vital area after the first bomb has breached a security barrier. Also, this threat envisions a small attacking force equipped with light weapons, rather than a larger force (e.g., 20 persons) equipped with heavier weapons such as anti-tank missiles. In sum, the present DBT is inadequate in light of the present threat environment. The compensatory measures required by the NRC's recent orders do not correct this deficiency.[42]

3. The Potential for Attacks on Nuclear Facilities

3.1 A Brief History

There is a rich history of events which show that acts of malice or insanity pose a significant threat to nuclear facilities around the world.[43] Consider some examples. Nuclear power plants under construction in Iran were repeatedly bombed from the air by Iraq in the period 1984-1987. Yugoslav Air Force fighters made a threatening overpass of the Krsko nuclear plant in Slovenia -- which was operating at the time -- a few days after Slovenia declared independence in 1991. So-called research reactors in Iraq were destroyed by aerial bombing by Israel in 1981 and by the United States in 1991. In 1987, Iranian radio threatened an attack by unspecified means on US nuclear plants if the United States attacked launch sites for Iran's Silkworm anti-ship missiles. Bombs damaged reactors under construction in Spain in

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[41] An NRC Fact Sheet (NRC, 2002) mentions new measures "against water-borne attacks", but it does not appear that these measures provide significant protection against boat bombs.
[42] POGO, 2002.
[43] Thompson, 1996.

1977 and in South Africa in 1982. Anti-tank missiles struck a nuclear plant under construction in France in 1982. North Korean commandos were killed while attempting to come ashore near a South Korean plant in 1985. These and other events illustrate the "external" threat to nuclear plants. Numerous crimes and acts of sabotage by plant personnel illustrate the "internal" threat.

Vehicle Bombs

The threat posed to nuclear facilities by vehicle bombs became clearly apparent from an October 1983 attack on a US Marine barracks in Beirut. In a suicide mission, a truck was driven at high speed past a guard post and into the barracks. A gas-boosted bomb on the truck was detonated with a yield equivalent to about 5 tonnes of TNT, destroying the building and killing 241 Marines. In April 1984 a study by Sandia National Laboratories titled "Analysis of Truck Bomb Threats at Nuclear Facilities" was presented to the NRC. According to an NRC summary:[44] "The results show that unacceptable damage to vital reactor systems could occur from a relatively small charge at close distances and also from larger but still reasonable size charges at large setback distances (greater than the protected area for most plants)." Eventually, in 1994, the NRC introduced regulations that require reactor licensees to install defenses (gates, barriers, etc.) against vehicle bombs. The NRC was spurred into taking this action by two incidents in February 1993. In one incident, a vehicle bomb was detonated in a parking garage under the World Trade Center in New York. In the other incident, a man recently released from a mental hospital crashed his station wagon through the security gate of the Three Mile Island nuclear plant and rammed the vehicle under a partly-opened door in the turbine building.

Suicidal Aircraft Attack

The threat of suicidal aircraft attack on symbolic or high-value targets became clearly apparent from three incidents in 1994.[45] In April 1994 a Federal Express flight engineer who was facing a disciplinary hearing was travelling as a passenger on a company DC-10. He stormed the cockpit, severely wounded all three members of the crew with a hammer, and tried to gain control of the aircraft. The crew regained control with great difficulty. Federal Express employees said that the flight engineer was planning to crash into a company building. In September 1994 a lone pilot crashed a stolen single-engine Cessna into the grounds of the White House, just short of the President's living quarters. In December 1994 four Algerians hijacked an Air France Airbus 300, carrying 20 sticks of dynamite. The aircraft landed in

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44 Rehm, 1984.

45 Wald, 2001.

Marseille, where the hijackers demanded that it be given a large fuel load -- three times more than necessary for the journey -- before flying to Paris. Troops killed the hijackers before this plan could be implemented. French authorities determined that the hijackers planned to explode the aircraft over Paris or crash it into the Eiffel Tower.

The Insider Threat

The incident involving the Federal Express flight engineer illustrates the vulnerability of industrial systems, including nuclear plants, to "internal" threats. That vulnerability is further illustrated by a number of incidents. In December 2000, Michael McDermott killed seven co-workers in a shooting rampage at an office building in Massachusetts. He had worked at the Maine Yankee nuclear plant from 1982 to 1988 as an auxiliary operator and operator before being terminated for exhibiting unstable behavior.[46] In 1997, Carl Drega of New Hampshire stockpiled weapons and killed four people -- including two state troopers and a judge -- on a suicide mission. He had passed security clearances at three nuclear plants in the 1990s.[47] In October 2000 a former US Army sergeant pleaded guilty to assisting Osama bin Laden in planning the bombing of the US embassy in Nairobi, which occurred in 1998.[48] In June 1999, a security guard at the Bradwell nuclear plant in Britain hacked into the plant's computer system and wiped out records. It emerged that he had never been vetted and had two undisclosed criminal convictions.[49] These and other incidents demonstrate clearly that it is foolish to ignore or downplay the "internal" threat of acts of malice or insanity at nuclear plants.

The General Threat of Terrorism

The events mentioned in the preceding paragraphs occurred against a background of numerous acts of terrorism around the world. Many of these acts have been highly destructive. US facilities have been targets on many occasions, as illustrated by the bombing of the US embassy in Beirut in 1983, the embassies in Nairobi and Dar es Salaam in 1998, and the USS Cole in 2000. There have been repeated warnings that the threat of terrorism is growing and could involve the US homeland. For example, in 1998 three authors with high-level government experience wrote:[50]

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[46] Barnard and Kerber, 2001.
[47] Ibid.
[48] Goldman, 2000.
[49] Maguire, 2001.
[50] Carter et al, 1998.

> "Long part of the Hollywood and Tom Clancy repertory of nightmarish scenarios, catastrophic terrorism has moved from far-fetched horror to a contingency that could happen next month. Although the United States still takes conventional terrorism seriously, as demonstrated by the response to the attacks on its embassies in Kenya and Tanzania in August, it is not yet prepared for the new threat of catastrophic terrorism."

Some years ago the US Department of Defense established an advisory commission on national security in the 21st century. This commission -- often known as the Hart-Rudman commission because it was co-chaired by former Senators Gary Hart and Warren Rudman -- issued reports in September 1999, April 2000 and March 2001. The findings in the September 1999 report included the following:[51]

> "America will become increasingly vulnerable to hostile attack on our homeland, and our military superiority will not entirely protect us.............States, terrorists and other disaffected groups will acquire weapons of mass destruction and mass disruption, and some will use them. Americans will likely die on American soil, possibly in large numbers."

It is clear that the potential for acts of malice or insanity at nuclear facilities -- including highly destructive acts -- has been foreseeable for many years, and has been foreseen. However, the terrorist attacks on the World Trade Center and the Pentagon on 11 September 2001 provided significant new information. These attacks conclusively demonstrated that the threat of highly-destructive acts of malice or insanity is a clear and present danger, and that no reasonable person can regard this threat as remote or speculative. According to press reports, US authorities have obtained information suggesting that the hijackers of United Airlines flight 93, which crashed in Pennsylvania on 11 September 2001, were planning to hit a nuclear plant.[52] This may be true or false, or the truth may never be known. Whatever the truth is, it would be foolish to regard nuclear plants as immune from attack.

Estimating the Probability of an Attack on a Nuclear Facility

The NRC has a longstanding policy of dismissing citizens' concerns about nuclear-facility accidents if the probability of such accidents is, in the agency's judgement, low. A body of analytic techniques known as probabilistic risk

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[51] Commission on National Security, 1999.

[52] Rufford et al, 2001.

assessment (PRA) has been developed to support such judgements.[53] However, the NRC Staff has conceded that it cannot provide a quantitative assessment of the probability of an act of malice at a nuclear facility. In a memo to the NRC Commissioners, the Staff has stated:[54]

> "The staff, as a result of its ongoing work with the Federal national security agencies, has determined that the ability to quantify the likelihood of sabotage events at nuclear power plants is not currently supported by the state-of-the-art in PRA methods and data. The staff also believes that both the NRC and the other government stakeholders would need to conduct additional research and expend significant time and resources before it could even attempt to quantify the likelihood of sabotage events. In addition, the national security agencies, Intelligence Community, and Law Enforcement Agencies do not currently quantitatively assess the likelihood of terrorist, criminal, or other malevolent acts."

To date, there has been no determined attack on a US civilian nuclear facility. At present, we cannot quantitatively estimate the probability of such an attack in the future. However, from a qualitative perspective, it is clear that the probability is significant.

3.2 The Strategic Context

In considering the need to defend civilian nuclear facilities, one is obliged to take a broad view of the security environment. An ISFSI, for example, may remain in service for 100 years or more. During that period the level of risk will vary but the cumulative risk will continue to grow. Thus, the ISFSI's designer should take a conservative position in specifying a DBT. That position should be informed by a sober assessment of the range of threats that may be manifested over coming decades.

A Turbulent World?

A number of strategic analysts have warned that world affairs may become more turbulent over the coming decades. Analysts have pointed to destabilizing factors that include economic inequality, poverty, political grievances, nationalism, environmental degradation and the weakening of international institutions. For example, a 1995 RAND study for the US Department of Defense contains the statement:[55]

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53 The state of the art of PRA can be illustrated by: NRC, 1990. For a critique of PRA, see: Hirsch et al, 1989.

54 Travers, 2001.

55 Kugler, 1995, page xv.

> "If the worst does transpire, the world could combine the negative features of nineteenth-century geopolitics, twentieth-century political passions, and twenty-first century technology: a chronically turbulent world of unstable multi-polarity, atavistic nationalism, and modern armaments."

As another example, the Stockholm Environment Institute (SEI) has identified a range of scenarios for the future of the world over the coming decades, and has studied the policies and actions that will tend to make each scenario come true. In summarizing this work, SEI states:[56]

> "In the critical years ahead, if destabilizing social, political and environmental stresses are addressed, the dream of a culturally rich, inclusive and sustainable world civilization becomes plausible. If they are not, the nightmare of an impoverished, mean and destructive future looms. The rapidity of the planetary transition increases the urgency for vision and action lest we cross thresholds that irreversibly reduce options -- a climate discontinuity, locking-in to unsustainable technological choices, and the loss of cultural and biological diversity."

SEI has specifically considered the implications of the September 2001 terrorist attacks, concluding:[57]

> "Certainly the world will not be the same after 9/11, but the ultimate implications are indeterminate. One possibility is hopeful: new strategic alliances could be a platform for new multinational engagement on a wide range of political, social and environmental problems. Heightened awareness of global inequities and dangers could support a push for a more equitable form of global development as both a moral and a security imperative. Popular values could eventually shift toward a strong desire for participation, cooperation and global understanding. Another possibility is ominous: an escalating spiral of violence and reaction could amplify cultural and political schisms; the new military and security priorities could weaken democratic institutions, civil liberties and economic opportunity; and people could grow more fearful, intolerant and xenophobic as elites withdraw to their fortresses."

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56 Raskin et al, 2002, page 11.
57 Ibid.

Nuclear Facilities as Symbolic Targets

In view of the range of possibilities for world order or turbulence over the coming decades, it would be prudent to assume that any US civilian nuclear facility could be the subject of a determined attack. Moreover, civilian nuclear facilities may be especially prime targets because of their symbolic connection with nuclear weapons. The US government flaunts its superiority in nuclear weapons and rejects any constraint on these weapons through international law.[58] At the same time, the government has signaled its willingness to attack Iraq because that country might acquire a nuclear weapon. It would be prudent to assume that this situation will motivate terrorist groups to search for ways to attack US nuclear facilities. For example, a terrorist group possessing a crude nuclear weapon might choose to use that weapon on a US civilian nuclear facility for two reasons. First, because the target would be highly symbolic. Second, because the radioactive fallout from the weapon would be greatly amplified.

The Domestic Threat

There is a natural tendency to look outside the country for sources of threat. However, an attack on a nuclear facility could also originate within the United States. The national strategy for homeland security contains the statement:[59]

> "Terrorist groups also include domestic organizations. The 1995 bombing of the Murrah Federal Building in Oklahoma City highlights the threat of domestic terrorist acts designed to achieve mass casualties. The US government averted seven planned terrorist acts in 1999 -- two were potentially large-scale, high-casualty attacks being organized by domestic extremist groups."

3.3 The US Government's Response to this Threat

The preceding discussion shows that there is a significant potential for a determined attack on a US civilian nuclear facility. Such an attack could employ a level of sophistication and violence that is characteristic of military operations. However, in most attack scenarios the attacking group would have a negligible capability for direct confrontation with US military forces. Thus, it is appropriate to think of an attack of this kind as a form of asymmetric warfare. The attacking group, be it domestic or foreign, will have

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58 Deller, 2002; Scarry, 2002.
59 Office of Homeland Security, 2002, page 10.

a set of political objectives. For symbolic and practical reasons, the attackers will prefer to obtain their weapons and logistical resources inside the USA.

US Strategy for National Security and Homeland Security

The White House has recently articulated a national security strategy for the United States.[60] This strategy rests primarily on the use of military force outside the country, to deter, disrupt or punish potential attackers. In support of this concept, the strategy asserts the right to conduct unilateral, pre-emptive attacks around the world, and repudiates the International Criminal Court. Homeland security is regarded as a secondary form of defense, as illustrated by the statement:[61]

> "While we recognize that our best defense is a good offense, we are also strengthening America's homeland security to protect against and deter attack."

A strategy for homeland security has been articulated by the White House.[62] This document contains a section titled "Defending against Catastrophic Threats", and that section begins with an aerial photograph of a nuclear power plant. Yet, the section does not mention civilian nuclear facilities or the NRC. Thus, at the highest levels of strategic planning, the US government has nothing to say about the potential for an attack on a nuclear facility, or about the measures that could be taken to defend against such attacks. In fact, the US government seems largely unaware of this threat, and has delegated its responsibility to the NRC. As described in Section 2.3 of this report, the NRC's response to the threat has been limited and ineffectual.

Imbalance in National Security and Defense Planning

Inattention to the vulnerability of nuclear facilities is symptomatic of a larger imbalance in national security and defense planning. As another example of imbalance, consider the threat of attack on the United States by inter-continental ballistic missiles (ICBMs). Large expenditures are devoted to the development of technologies that might, ultimately, allow missile warheads to be intercepted. Yet, in considering the respective risks of attack by missiles or other means, the US National Intelligence Council has stated:[63]

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[60] White House, 2002.
[61] Ibid, page 6.
[62] Office of Homeland Security, 2002.
[63] National Intelligence Council, 2001, page 18.

> "Nonmissile means of delivering weapons of mass destruction [WMD] do not provide the same prestige or degree of deterrence and coercive diplomacy associated with ICBMs. Nevertheless, concern remains about options for delivering WMD to the United States without missiles by state and nonstate actors. Ships, trucks, airplanes, and other means may be used. In fact, the Intelligence Community judges that US territory is more likely to be attacked with WMD using nonmissile means, primarily because such means:
>
> - Are less expensive than developing and producing ICBMs.
> - Can be covertly developed and employed; the source of the weapon could be masked in an attempt to evade retaliation.
> - Probably would be more reliable than ICBMs that have not completed rigorous testing and validation programs.
> - Probably would be much more accurate than emerging ICBMs over the next 15 years.
> - Probably would be more effective for disseminating biological warfare agent than a ballistic missile.
> - Would avoid missile defenses."

The defense analyst John Newhouse has contrasted the high level of attention given to the ICBM threat with the lack of effort in other areas of defense. He notes that the State Department advised US embassies in early 2001 that the principal threat to US security is the use of long-range missiles by rogue states, and comments:[64]

> "This dubious proposition -- an article of faith within parts of the defense establishment -- obscured existing and far more credible threats from truly frightful weapons, some of which are within the reach of terrorists. They include Russia's shaky control of its nuclear weapons and weapons-usable material; the vulnerability of US coastal cities and military forces stationed abroad to medium-range missile systems, ballistic and cruise; the vulnerabilities of all cities to chemical and biological weapons, along with so-called suitcase weapons and other low-tech delivery expedients. Vehicles that contain potentially destructive amounts of stored energy are a major source of concern, as is one of their most attractive potential targets, a nuclear spent-fuel storage facility."

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[64] Newhouse, 2002, page 43.

Nuclear Facilities as Targets

It is clear that US civilian nuclear facilities are candidates for attack under conditions of asymmetric warfare. They are large, fixed targets that are, at present, lightly defended. In the eyes of an enemy, they can be regarded as pre-deployed radiological weapons. They can be attacked using comparatively low levels of technology. Given the United States' overt reliance on nuclear weapons as offensive instruments, civilian nuclear facilities offer highly symbolic targets. In light of these considerations, it is remarkable that the US government has largely ignored this threat.

The Danger of an Offense-Dominated Strategy

At present, US policy for national security assigns a higher priority to offensive actions worldwide than to defensive actions within the homeland. This is a tradition of many years' standing. However, in the contemporary era of asymmetric warfare, this policy can be dangerous.[65] If our vulnerable infrastructure -- including nuclear facilities, the airlines, etc. -- is poorly defended, we may feel compelled to use military force aggressively around the world, in order to pre-empt or punish attackers. Such action poses the risk of arousing hostility and promoting anarchy, leading to new attacks on our homeland. The potential exists for an escalating spiral of violence. Strategic analysts have warned of this danger, both before and after the terrorist events of September 2001.[66]

3.4 A Balanced Response to the Threat

The United States needs a balanced, mature strategy for national defense and international security. Within that strategy, it needs a balanced strategy for homeland security. Finally, as a part of homeland security, the nation needs a defense-in-depth strategy to protect its civilian nuclear facilities. At present, all three levels of strategy are deficient.

The Role of Protection in a Balanced Response

Articulation of a balanced strategy at all three levels is a task beyond the scope of this report. However, this report does articulate, in Sections 4.4 and 4.5

[65] A recent essay (Betts, 2003) argues that US decision makers have neglected the risk that Iraq's leaders will strike back at the US homeland if we attack Iraq. Betts' essay focusses on the potential for Iraq to use chemical or biological weapons on US territory, but the same general arguments apply to the potential for an attack on a US civilian nuclear facility.
[66] See, for example: Sloan, 1995; Martin, 2002 (see especially the chapter by Conrad Crane in this volume); Mathews, 2002; Conetta, 2002; Crawford, 2003; and Newhouse, 2002.

respectively, a defense-in-depth strategy for nuclear facilities and a national strategy for robust storage of spent fuel. As an illustration of how these protective measures could fit within a higher-level strategy, consider Carl Conetta's suggestion of a four-pronged campaign against the terrorist group al-Qaeda. The four prongs would be:[67]

> "(i) squeeze the blood flow of the organization -- its financial support system;
> (ii) throw more light on the organization's members and components through intelligence gathering activities;
> (iii) impede the movement of the organization by increasing the sensitivity of screening procedures at critical gateways -- borders, financial exchanges, arms markets, and transportation portals; and
> (iv) improve the protection of high-value targets."

The importance of protecting high-value targets is emphasized in the recent report of a high-level task force convened by the Council on Foreign Relations and chaired by former Senators Gary Hart and Warren Rudman. One of the report's major findings is:[68]

> "*Homeland security measures have deterrence value:* US counterterrorism initiatives abroad can be reinforced by making the US homeland a less tempting target. We can transform the calculations of would-be terrorists by elevating the risk that (1) an attack on the United States will fail, and (2) the disruptive consequences of a successful attack will be minimal. It is especially critical that we bolster this deterrent now since an inevitable consequence of the US government's stepped-up military and diplomatic exertions will be to elevate the incentive to strike back before these efforts have their desired effect."

The Need for Proactive Planning

Other findings by the Council on Foreign Relations' task force also deserve attention. For example, their report points out that proactive planning will yield better protection at lower cost than reacting after each new attack.[69] This point is especially important in an era of asymmetric warfare, when opponents will employ unfamiliar tactics. Planning techniques such as "competitive strategies" and "net assessment" have been developed to accommodate such situations. In discussing net assessment, one author has stated:[70]

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67 Conetta, 2002, page 3.
68 Hart et al, 2002, pp 14-15.
69 Ibid, page 16.
70 Hoffman, 2002, pp 3-4.

> "One of the advantages of such an approach is that it credits the opponent with having a brain and a will, which Clausewitz suggested is also fundamental to war. Rarely do US strategists credit adversaries with being as cunning or adaptive as they usually turn out to be. It is well to be reminded on occasion that any opponent has strategies and options at his disposal too. The essence of the homeland security challenge is based on this consideration."

4. Defending Nuclear Power Plants and Spent Fuel

4.1 Potential Modes and Instruments of Attack

It is not appropriate to publish a detailed discussion of scenarios whereby a nuclear power plant or a spent-fuel-storage facility might be successfully attacked. However, it must be assumed that attackers are technically sophisticated and possess considerable knowledge about individual nuclear facilities. For decades, engineering drawings, photographs and technical analyses have been openly available for every civilian nuclear facility in the USA. This material is archived at many locations around the world. Thus, a public discussion, in general terms, of potential modes and instruments of attack will not assist attackers. Indeed, such a discussion is needed to ensure that appropriate defensive actions are taken.

Safety Systems and their Vulnerability

The safe operation of a US commercial reactor or a spent-fuel pool depends upon the fuel in the reactor or the pool being immersed in water. Moreover, that water must be continually cooled to remove fission heat or radioactive decay heat generated in the fuel. A variety of systems are used to ensure that water is available and is cooled, and that other safety-related functions -- such as shutdown of the fission reaction when needed -- are performed. Some of the relevant systems -- such as the switchyard -- are highly vulnerable to attack. Other systems are located inside reinforced-concrete structures -- such as the reactor auxiliary building -- that provide some degree of protection against attack. The reactor itself is inside a containment structure. At some plants, but not all, the reactor containment is a concrete structure that is highly reinforced and comparatively robust. Spent-fuel pools have thick concrete walls but are typically covered by lightweight structures.

Attack through Brute Force or Indirectly?

A group of attackers equipped with highly-destructive instruments could take a brute-force approach to attacking a reactor or a spent-fuel pool. Such an approach would aim to directly breach the reactor containment and primary cooling circuit, or to breach the wall or floor of a spent-fuel pool. Alternatively, the attacking group could take an indirect approach, and many such approaches will readily suggest themselves to technically-informed attackers. Insiders, or outsiders who have taken over the plant, could obtain a release of radioactive material without necessarily employing destructive instruments. Some attack scenarios will involve the disabling of plant personnel, which could be accomplished by armed attack, use of lethal chemical weapons, or radioactive contamination of the site by an initial release of radioactive material.

Vulnerability of ISFSIs

Dry-storage ISFSIs differ from reactors and spent-fuel pools in that their operation is entirely passive. Thus, each dry-storage module in an ISFSI must be attacked directly. To obtain a release of radioactive material, the wall of the fuel container must be penetrated from the outside, or the container must be heated by an external fire to such an extent that the containment envelope fails. The attack could also exploit stored chemical energy in the zirconium cladding of spent fuel inside the module. Combustion of this cladding in air, if initiated, would generate heat that could liberate radioactive material from the fuel to the outside environment. A knowledgeable attacker could combine penetration of the fuel container with the initiation of combustion.

Relevance of Site-Security Barriers

In some attack scenarios that involve the use of destructive instruments, the attackers may need to carry these instruments, by hand or in a vehicle, to the point of application. Such an action would require the overcoming of site-security barriers. In other scenarios, the instruments could be launched from a position outside some or all of these barriers.

Commercial Aircraft as Instruments of Attack

One instrument that an attacking group will consider is a fuel-laden commercial aircraft. As indicators of the forces that could accompany the impact of such an aircraft, consider the weights and fuel capacities of some typical jetliners.[71] The Boeing 737-300 has a maximum takeoff weight of 56-63 tonnes and a fuel capacity of 20-24 thousand liters. The Boeing 747-400 has a maximum takeoff weight of 363-395 tonnes and a fuel capacity of 204-217 thousand liters. The Boeing 757 has a maximum takeoff weight of 104-116 tonnes and a fuel capacity of 43 thousand liters. The Boeing 767 has a

[71] Jackson, 1996.

maximum takeoff weight of 136-181 tonnes and a fuel capacity of 63-91 thousand liters.

Commercial jet fuel typically has a heat of combustion of about 38 MJ per liter. For comparison, 1 kilogram of TNT will yield 4.2 MJ of energy. Thus, complete combustion of 1 liter of jet fuel will yield energy equivalent to that from 9 kilograms of TNT. Complete combustion of 100 thousand liters of jet fuel -- about half the fuel capacity of a Boeing 747-400 -- will yield energy equivalent to that from 900 tonnes of TNT. Thus, the impact of a fuel-laden aircraft could lead to a violent fuel-air explosion. Fuel-air munitions have been developed that yield more than 5 times the energy of their equivalent weight in TNT, and create a blast overpressure exceeding 1,000 pounds per square inch.[72] A fuel-air explosion arising from an aircraft impact will be less efficient than a munition in converting combustion energy into blast, but could generate a highly-destructive blast if fuel vapor accumulates in a confined space before igniting.

Explosive-Laden, General-Aviation Aircraft

The attacking group might choose to use an explosive-laden, general-aviation aircraft as an instrument of attack. Such an aircraft could be much easier to obtain than a large commercial aircraft. In 1999, about 219,000 general-aviation aircraft were in use in the USA.[73] Of these, about 172,000 had piston engines, 5,700 were turboprops, 7,100 were turbojets and 7,400 were helicopters.[74] In the piston-engine category, about 21,000 aircraft had two engines, the remainder having one engine. The general-aviation fleet in 2002 must be similar to that in 1999.

It is clear that terrorist groups can readily obtain and use explosive materials. Such use is a tragic accompaniment to political disputes around the world. Moreover, explosives are easily obtainable within the USA. In 2001, about 2.4 million tonnes of explosives were sold in the USA. Most of this material consisted of blasting agents and oxidizers used for mining, quarrying and construction. Much of the blasting material consisted of mixtures of ammonium nitrate and fuel oil, which are readily-available materials. It is also noteworthy that current law in many US states allows high explosives to be purchased without a permit or a background check.[75]

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[72] Gervasi, 1977.

[73] Data from the website of the General Aviation Manufacturers Association (www.generalaviation.org), 30 September 2002.

[74] The remainder of the fleet consisted of gliders, balloons/blimps and experimental aircraft.

[75] Information from the website of the Institute of Makers of Explosives (www.ime.org), 30 September 2002.

Anti-Tank Missiles

Another instrument of attack that could be used is an anti-tank missile. Large numbers of these missiles exist around the world, and they can be obtained by many terrorist groups. As an example, consider the tube-launched, optically-tracked, wire-guided (TOW) anti-tank missile system, which is now marketed by Raytheon.[76] This system is said to be the most successful anti-tank missile system in the world. It first entered service with the US Army in 1970 and is currently in use by more than 40 military forces. As of 1991, more than 460,000 TOW missiles had been produced, and the cumulative production up to 2002 must be substantially higher. The TOW missile has a maintenance-free storage life of 20 years, and all versions of the missile can be fired from any TOW launcher. TOW systems have been sold to countries such as Colombia, Iran, Lebanon, Pakistan, Somalia, Yugoslavia and South Yemen, so it must be presumed that they can be obtained by terrorist groups. There is no indication from available literature that the TOW missile or launcher is self-disabling if it passes into inappropriate hands. In connection with the availability of systems of this kind, it is interesting to note that, in August 2002, federal agents seized more than 2,300 unregistered armor-piercing missiles from a private, counter-terrorism training school in New Mexico.[77]

Modern anti-tank missiles are reliable, accurate and easy to use. They are capable of penetrating considerable thicknesses of armor plate using a shaped-charge warhead that is designed for this purpose. Some types of missile can also be equipped with a general-purpose warhead that would be used for attacking targets such as fortified bunkers and gun emplacements. All types can be set up and reloaded comparatively quickly. Consider the TOW missile system as an example. The launcher can be mounted on a light vehicle or carried a short distance and mounted on the ground on a tripod. A late-model TOW launcher weighs about 93 kilograms, the guidance set about 24 kilograms and each missile about 22 kilograms. A rate of fire of about two rounds per minute can be sustained, and the missile has a range in excess of 3,000 meters. It is reported that an early-model TOW missile can blow a hole as much as two feet in diameter in the armor of a Soviet T-62 tank, or cut through three feet of concrete. Later-model TOW missiles are more capable.[78]

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[76] Information from: Hogg, 1991; Gervasi, 1977; Raytheon website (www.raytheon.com), 26 September 2002; US Marine Corps website (www.hqmc.usmc.mil), 26 September 2002; and Canadian Army website (www.army.forces.gc.ca), 27 September 2002.

[77] Reuters, 2002.

[78] Information from: Hogg, 1991; Gervasi, 1977; Raytheon website (www.raytheon.com), 26 September 2002; US Marine Corps website (www.hqmc.usmc.mil), 26 September 2002; and Canadian Army website (www.army.forces.gc.ca), 27 September 2002.

Nuclear Weapons

A nuclear weapon could be used to attack a civilian nuclear facility. This possibility was a source of concern during the Cold War, and there is a body of technical and policy literature on this subject.[79] Russia retains the capability to attack US nuclear facilities using ICBMs with thermonuclear warheads, and might be motivated at some future date to threaten or implement such an attack. A greater concern in the current period is that a sub-national group, with or without the assistance of a government, might use a comparatively low-yield fission weapon -- perhaps one with an explosive yield in the vicinity of 10 kilotonnes of TNT equivalent -- to attack a US nuclear facility. The means of delivery might be a land vehicle or a general-aviation aircraft. Such a weapon would be difficult to obtain, but many knowledgeable experts have warned that the fissionable material for a simple nuclear weapon could be diverted from poorly-secured stocks in Russia and elsewhere.[80] There is even the possibility that a complete nuclear weapon will be diverted. A high-level group advising the US government has examined the security of nuclear weapons and fissile material in Russia, concluding:[81]

> "The most urgent unmet national security threat to the United States today is the danger that weapons of mass destruction or weapons-usable material in Russia could be stolen and sold to terrorists or hostile nation states and used against American troops abroad or citizens at home. This threat is a clear and present danger to the international community as well as to American lives and liberties."

Summary

Table 1, on the following page, briefly summarizes the characteristics of some potential modes of attack on civilian nuclear facilities, and the present defense against each mode. Other modes of attack can be identified, and an attacking group might use several modes simultaneously or in sequence. The characteristics of each mode are, of course, more complex and varied than is shown in Table 1. Nevertheless, this table shows that determined, knowledgeable attackers have a range of options available to them.

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[79] See, for example: Fetter, 1982; Fetter and Tsipis, 1980; Ramberg, 1984; and SIPRI, 1981.
[80] See, for example: Baker, Cutler et al, 2001; Bunn et al, 2002; and Sokolski and Riisager, 2002.
[81] Baker, Cutler et al, 2001, first page of Executive Summary.

MODE OF ATTACK	CHARACTERISTICS	PRESENT DEFENSE
Commando-style attack	• Could involve heavy weapons and sophisticated tactics • Successful attack would require substantial planning and resources	Alarms, fences and lightly-armed guards, with offsite backup
Land-vehicle bomb	• Readily obtainable • Highly destructive if detonated at target	Vehicle barriers at entry points to Protected Area
Anti-tank missile	• Readily obtainable • Highly destructive at point of impact	None if missile launched from offsite
Commercial aircraft	• More difficult to obtain than pre-9/11 • Can destroy larger, softer targets	None
Explosive-laden smaller aircraft	• Readily obtainable • Can destroy smaller, harder targets	None
10-kilotonne nuclear weapon	• Difficult to obtain • Assured destruction if detonated at target	None

TABLE 1

SOME POTENTIAL MODES OF ATTACK ON CIVILIAN NUCLEAR FACILITIES

4.2 Vulnerability to Attack

The preceding section of this report describes, in deliberately general terms, the potential modes and instruments of attack on a nuclear power plant or an ISFSI. No sensitive information is disclosed. In discussing the vulnerability of nuclear facilities to such attacks, one must be similarly careful to avoid disclosing sensitive information. In this context, the word "vulnerability" refers to the potential for an act of malice or insanity to cause a release of radioactive material to the environment. At the site of a nuclear power plant or an ISFSI, most of the radioactive material at the site is in the reactor(s), the spent-fuel pool(s) and the ISFSI modules.

Requirements for a Vulnerability Study

Every US commercial reactor has been subjected to a PRA-type study by the licensee. This study addressed the reactor's potential to experience accidents, but did not consider acts of malice or insanity. No spent-fuel pool or ISFSI has been subjected to a PRA-type study or a study of its vulnerability to acts of malice or insanity. Indeed, there has never been a comprehensive, published study of the vulnerability of any US nuclear facility to acts of malice or insanity. Spurred by the terrorist events of September 2001, the NRC has sponsored secret, ongoing studies on the vulnerability of nuclear facilities to impact by a large commercial aircraft. Available information suggests that these studies are narrow in scope and will provide limited guidance regarding the overall vulnerability of nuclear facilities.

A comprehensive study of a facility's vulnerability would begin by identifying a range of potential attacks on the facility. The probability of each potential attack would be qualitatively estimated, with consideration of the factors (e.g., international events, changing availability of instruments of attack) that could alter the probability over time. Site-specific factors affecting the feasibility and probability of attack scenarios include local terrain and the proximity of coastlines, airports, population centers and national symbols. A variety of modes and instruments of attack would be considered, of the kind discussed in Section 4.1.

After identifying a range of potential attacks, a comprehensive study would examine the vulnerability of the subject facility to those attacks. This could be done by adapting and extending known techniques of PRA, with an emphasis on the logical structure of PRA rather than the numerical probabilities of events. The analysis would consider the potential for interactions among facilities at a site. For example, a potentially important interaction could be the prevention of personnel access at one facility (e.g., a spent-fuel pool) due to a release of radioactive material at another facility (e.g., a reactor). Attention would be given to the potential for "cascading" scenarios in which attacks at some parts of a nuclear-power-plant site (e.g., control room, switchyard, diesel generators) lead to releases from reactors and/or spent fuel pools that were not directly attacked.

Working with Partial or Misleading Information

In the absence of any comprehensive study of vulnerability, one is obliged to rely upon partial information. Also, one must contend with misleading information disseminated by the nuclear industry. An egregious example is a recent paper in Science, a journal that is usually sound.[82] Two points illustrate the low scientific quality of this paper. First, the paper cites an experiment performed at Sandia National Laboratories as proof that an aircraft cannot penetrate the concrete wall of a reactor containment. In response, Sandia officials have stated that the test has no relevance to the structural behavior of a containment wall, a fact that is readily evident from the nature of the test.[83] Second, the paper states, in connection with the vulnerability of spent-fuel shipping casks, that "there is virtually nothing one could do to these shipping casks that would cause a significant public hazard".[84] A report prepared by Sandia for the NRC is cited in support of this statement.[85] Yet, examination of the Sandia report reveals that it considers only the effects on a shipping cask of impact and fire pursuant to a truck or train accident. The Sandia report does not address the effects of, for example, attack by a TOW missile, a vehicle bomb, or a manually-placed charge.

Aircraft Impact

A rough, preliminary indication of the vulnerability of a nuclear power plant to aircraft impact can be obtained from the PRA for the Seabrook reactor. This reactor is a 4-loop Westinghouse PWR with a large, dry containment, and is one of only four US reactors that were specifically designed to resist impact by an aircraft, a 6-tonne aircraft in the case of Seabrook.[86] The Seabrook PRA finds that any direct impact on the containment by an aircraft weighing more than 37 tonnes will lead to penetration of the containment and a breach in the reactor coolant circuit. Also, the Seabrook PRA finds that a similar impact on the control building or auxiliary building will inevitably lead to a core melt.[87] All of the typical, commercial aircraft mentioned in Section 4.1 of this

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82 Chapin et al, 2002.
83 Jones, 2002a.
84 Chapin et al, 2002, page 1997.
85 Sprung et al, 2000.
86 Markey, 2002, page 73.
87 PLG, 1983, pp 9.3-10 to 9.3-11.

report weigh considerably more than 37 tonnes. Moreover, the Seabrook PRA does not consider the effects of a fuel-air explosion and/or fire as an accompaniment to an aircraft impact. Finally, this PRA, like other PRAs, does not consider malicious acts. Thus, it does not consider, for example, an attack on the Seabrook reactor by an explosive-laden, general-aviation aircraft.

Analytic techniques are available for estimating the effects that aircraft impact will have on the structures and equipment of a nuclear facility. Two recent studies illustrate the use of such techniques. First, the Nuclear Energy Institute (NEI), an industry lobbying organization, has released some preliminary findings from an analysis of aircraft impact on reactor containments and spent-fuel facilities.[88] The analysis itself will not be published, so the findings cannot be verified. Second, a group at Purdue University has released the results of its simulation of the aircraft impact on the Pentagon that occurred on 11 September 2001.[89] A simulation of this kind could be performed for aircraft impact on a nuclear facility. The Purdue group employs commercially-available software and, in contrast to NEI, seems willing to publish its analysis.

The analytic techniques discussed in the preceding paragraph focus on the kinetic energy of the impacting aircraft and its contents. Effects of an accompanying fuel-air explosion and/or fire are given, at best, a crude analysis. A 1982 review by Argonne National Laboratory of the state of the art for aircraft-impact analysis stated:[90]

> "Based on the review of past licensing experience, it appears that fire and explosion hazards have been treated with much less care than the direct aircraft impact and the resulting structural response. Therefore, the claim that these fire/explosion effects do not represent a threat to nuclear power plants has not been clearly demonstrated."

Examination of PRAs and related studies for nuclear facilities shows that the Argonne statement remains valid today. Indeed, in view of the large amount of energy that can be liberated in a fuel-air fire or explosion, previous analyses of aircraft impacts may have substantially underestimated the vulnerability of nuclear facilities to such impacts.

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[88] NEI, 2002.

[89] Purdue, 2002; Sozen et al, 2002.

[90] Kot et al, 1982, page 78.

Vulnerability of Spent-Fuel Pools

The vulnerability of spent-fuel pools deserves special attention because these pools contain large amounts of long-lived radioactive material that could be liberally released to the atmosphere during a fire.[91] The potential for such a fire exists because the pools have been equipped with high-density racks. In the 1970s, the spent-fuel pools of US nuclear power plants were typically equipped with low-density, open-frame racks. If water were partially or totally lost from such a pool, air or steam could circulate freely throughout the racks, providing convective cooling to the spent fuel. By contrast, the high-density racks that are used today have a closed structure. To suppress criticality, each fuel assembly is surrounded by solid, neutron-absorbing panels, and there is little or no gap between the panels of adjacent cells. In the absence of water, this configuration allows only one mode of circulation of air and steam around a fuel assembly -- vertically upward within the confines of the neutron-absorbing panels.

If water is totally lost from a high-density pool, air will pass downward through available gaps such as the gap between the pool wall and the outer faces of the racks, will travel horizontally across the base of the pool, will enter each rack cell through a hole in its base, and will rise upward within the cell, providing cooling to the spent fuel assembly in that cell. If the fuel has been discharged from the reactor comparatively recently, the flow of air may be insufficient to remove all of the fuel's decay heat. In that case, the temperature of the fuel cladding may rise to the point where a self-sustaining, exothermic oxidation reaction with air will begin. In simple terms, the fuel cladding -- which is made of zirconium alloy -- will begin to burn. The zirconium-alloy cladding can also enter into a self-sustaining, exothermic oxidation reaction with steam. Other exothermic oxidation reactions can also occur. For simplicity, the occurrence of one or more of the possible reactions can be referred to as a pool fire.

In many scenarios for loss of water from a pool, the flow of air that is described in the preceding paragraph will be blocked. For example, a falling object (e.g., a fuel-transfer cask) might distort rack structures, thereby blocking air flow. As another example, an attack might cause debris (e.g., from the roof of the fuel handling building) to fall into the pool and block air flow. The presence of residual water in the bottom of the pool would also block air flow. In most scenarios for loss of water, residual water will be present for significant periods of time. Blockage of air flow, for whatever reason, will lead to ignition of fuel that has been discharged from a reactor for long

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[91] The NRC has published a variety of technical documents that address spent-fuel-pool fires. The most recent of these documents is: Collins et al, 2000.

periods -- potentially 10 years or longer.[92] The NRC Staff failed to understand this point for more than two decades.[93]

Loss of Water from a Pool

Partial or total loss of water from a spent-fuel pool could occur through leakage, evaporation, siphoning, pumping, displacement by objects falling into the pool, or overturning of the pool. These modes of loss of water could arise, directly or indirectly, through a variety of attack scenarios. As a simple example, consider leakage as a direct result of aircraft impact on the wall of a pool. This example represents a brute-force attack on the model of 11 September 2001. Other attack options will suggest themselves to knowledgeable attackers.

An NRC Staff study includes a crude, generic analysis of the conditional probability that aircraft impact will cause a loss of water from a spent fuel pool.[94] The pool is assumed to have a 5-ft-thick reinforced-concrete wall. Impacting aircraft are divided into the categories "large" (weight more than 5.4 tonnes) and "small" (weight less than 5.4 tonnes). The Staff estimates that the conditional probability of penetration of the pool wall by a large aircraft is 0.45, and that 50 percent of penetration incidents involve a loss of water which exposes fuel to air. Thus, the Staff estimates that, for impact of a large aircraft, the conditional probability of a loss of water sufficient to initiate a pool fire is 0.23 (23 percent).

Facility Interactions and Cascading Scenarios

An earlier paragraph in Section 4.2 of this report mentions the potential for interactions among facilities on a site, and points out that a potentially important interaction could be the prevention of personnel access at one facility (e.g., a spent-fuel pool) due to a release of radioactive material at another facility (e.g., a reactor). This type of interaction was partially addressed during a licensing proceeding for the Harris nuclear power plant. In that proceeding, the NRC Staff conceded that a fire in one spent-fuel pool would preclude the provision of cooling and makeup to nearby pools, thereby leading to evaporation of water from the nearby pools followed by fires in those pools.[95] This situation would arise mostly because the initial fire would contaminate the site with radioactive material, generating high radiation fields that preclude personnel access. An analogous situation could

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[92] The role of residual water in promoting ignition of old fuel is discussed in: Thompson, 1999, Appendix D.
[93] Thompson, 2002a, Section II.
[94] Collins et al, 2000, page 3-23 and Appendix 2D.
[95] Parry et al, 2000, paragraph 29.

arise in which the release of radioactive material from a damaged reactor precludes the provision of cooling and makeup to nearby pools. For example, an attack on a reactor could lead to a rapid-onset core melt with an open containment, accompanied by a raging fire. That event would create high radiation fields across the site, potentially precluding any access to the site by personnel. One can envision a variety of "cascading" scenarios, in which there might eventually be fires in all of the pools at a site, accompanied by core-melt events at all of the reactors.

Progression of a Pool Fire

A pool fire could begin comparatively soon after water is lost from a pool. For example, suppose that most of the length of the fuel assemblies is exposed to air, but the flow of air to the base of the racks is precluded by residual water or a collapsed structure. In that event, fuel heatup would be approximately adiabatic. Fuel discharged for 1 month would ignite in less than 2 hours, and fuel discharged for 3 months would ignite in about 3 hours. The fire would then spread to older fuel. Once a fire has begun, it could be impossible to extinguish. Spraying water on the fire could feed an exothermic zirconium-steam reaction that would generate flammable hydrogen. High radiation fields could preclude the approach of firefighters.

Vulnerability of Dry-Storage Modules

The dry-storage modules used at ISFSIs are passively safe, as discussed in Section 4.1 of this report. Thus, an attacking group that seeks to obtain a radioactive release from an ISFSI must attack each module directly. To obtain a release of radioactive material, the wall of the fuel container must be penetrated from the outside, or the container must be heated by an external fire to such an extent that the containment envelope fails. In addition, a technically-informed and appropriately-equipped attacker could exploit stored chemical energy in the zirconium cladding of the stored spent fuel. Such an attacker would arrange for penetration of the container to be accompanied by the initiation of combustion of the cladding in air. Combustion would generate heat that could liberate radioactive material from the fuel to the outside environment. Initiation of combustion could be facilitated by the presence of zirconium hydride in the fuel cladding, which is a characteristic of high-burnup fuel. The NRC Staff has noted that zirconium hydride can experience auto-ignition in air.[96] This point had been brought to the Staff's attention by the NRC's Advisory Committee on Reactor Safeguards.[97]

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[96] Collins et al, 2000, page A1B-3.
[97] Powers, 2000, page 3.

There is a body of literature that addresses aspects of the vulnerability of dry-storage modules for ISFSIs. Consider some examples. First, NAC International has analyzed the impact of a Boeing 747-400 aircraft on a NAC-UMS storage module of the type discussed in Section 2.2 of this report.[98] According to NAC, this analysis shows that failure of the fuel container would not occur, either from impact or fire. Second, analyses of aircraft impact have been done in Germany in connection with the licensing of ISFSIs that employ CASTOR casks. In Germany, ISFSIs are typically located inside buildings to provide some protection against anti-tank missiles, a practice which creates the potential for pooling of jet fuel following an aircraft impact. As a result, the intensity and duration of fire has become a key issue in technical debates about the release of radioactive material following an aircraft impact.[99] Third, in a test done in Germany in 1992, a shortened CASTOR cask containing simulated fuel assemblies made from depleted uranium was penetrated by a static, shaped charge, in order to simulate attack by an anti-tank missile.[100] The metal jet created by the shaped charge caused a small amount of finely-divided uranium to be released from the cask, but this test did not account for several important factors that are discussed in the following paragraph. Fourth, analyses of aircraft, cruise-missile and dummy-bomb impact on a dry-storage module have been done in connection with the licensing of the proposed Skull Valley ISFSI. The accompanying technical debate suggests that the magnitude of the radioactive release following penetration of a fuel container would be sensitive to the fraction of a fuel assembly's inventory of radionuclides, such as cesium-137, that would be present in the pellet-cladding gap region.[101]

Requirements for a Comprehensive Study of Dry-Storage Vulnerability

The literature that is exemplified in the preceding paragraph addresses only some of the attack scenarios and physical-chemical phenomena that would be addressed in a comprehensive assessment of the vulnerability of dry-storage modules. Such an assessment would consider a range of instruments of attack, including anti-tank missiles, manually-placed charges, a vehicle bomb or an aircraft bomb. Modes of attack that promote zirconium ignition would be considered. Factors that would be accounted for include: (i) the presence of zirconium hydride in fuel cladding; (ii) radioactive-decay heat in fuel pellets; (iii) a pre-attack temperature characteristic of an actual, operating module; and (iv) source-term phenomena (such as the gap inventory of radionuclides) that are characteristic of high-burnup fuel. There is no evidence from

[98] McCough and Pennington, 2002.
[99] Hirsch, 2002.
[100] Lange et al, 2002.
[101] Resnikoff, 2001.

published literature that a comprehensive vulnerability assessment of this kind has been made. Some components of a comprehensive assessment may have been performed secretly. For example, there are rumors of NRC-sponsored tests that have combined penetration of a fuel container with incendiary effects. Given the information that is available, it is prudent to assume that a variety of modes and non-nuclear instruments of attack could release to the atmosphere a substantial fraction of the radioactive inventory of a dry-storage module.

Attack Using a Nuclear Weapon

As indicated in Section 4.1 of this report, it is prudent to assume that a low-yield nuclear weapon (with a yield of perhaps 10 kilotonnes of TNT equivalent) might be used as an instrument of attack at a nuclear power plant or an ISFSI. A thorough assessment of the vulnerability to such an attack of the reactor(s), spent-fuel pool(s) and ISFSI modules at a site would require detailed analysis. Absent such an analysis, rough judgements can be made.

Consider, for example, a 10-kilotonne ground burst at an unhardened, surface-level ISFSI of the usual US type. It seems reasonable to assume that any module within the crater area would, as a result of blast effects and heating by the fireball, become divided into fragments, many of them small enough to travel downwind for many kilometers before falling to earth. A 10-kilotonne ground burst over sandstone, which is perhaps representative of an ISFSI, would yield a crater about 68 meters in diameter and 16 meters deep.[102]

As an indication of the potential release of radioactive material following a nuclear detonation at an ISFSI, consider a 10-kilotonne groundburst at an ISFSI that employs vertical-axis fuel-storage modules with a center-to-center distance of 5.5 meters, as would be the case for the proposed Diablo Canyon facility. Given a large, square array of such modules, about 120 modules would fall within the 68-meter diameter of the anticipated crater. Thus, it is plausible to assume that 100 percent of the volatile radionuclides (such as cesium-137) in 120 modules, together with a lesser fraction of the non-volatile radionuclides, would be carried downwind in a radioactive plume. This quantity could be an over-estimate, because the ISFSI has finite dimensions and is not an infinite array, or it could be an under-estimate, because damage to modules outside the crater is not considered. Note that a NAC-UMS module, as used at Maine Yankee, can hold 24 PWR fuel assemblies or 56

[102] Glasstone, 1962, Chapter VI.

BWR fuel assemblies.[103] The HI-STORM 100 modules that would be used at the proposed Diablo Canyon ISFSI can each hold 32 PWR fuel assemblies.[104]

Comparative Risks of Attack Options

Section 4.1 of this report shows that a determined, knowledgeable group has available to it a range of options for attacking reactors, spent-fuel pools and ISFSIs. The preceding paragraphs of Section 4.2 provide a brief discussion of the vulnerability of reactors, pools and ISFSI modules to such options. These topics could be discussed more comprehensively, but that task was beyond the scope of this report. A comprehensive assessment -- whose underlying technical analysis should not, for obvious reasons, be openly published -- would identify a wide range of attack scenarios and would estimate their outcomes. Such an assessment could provide a perspective on the comparative risks of attack options.

As an illustration of comparative risk, consider three potential options for obtaining a release of radioactive material. Option I would be an attack on an ISFSI using a 10-kilotonne nuclear weapon delivered by a general-aviation aircraft. Delivery of the weapon could be straightforward, given the lack of air defense at ISFSIs, but the weapon would be difficult to obtain. Thus, this attack option seems to have a comparatively low probability. However, it would yield a large release of radioactive material. Option II would be a commando-style attack in which the attackers seize control of a nuclear power plant, initiate a reactor-core melt, breach the reactor containment, and initiate the removal of water from the spent-fuel pool(s) by siphoning and/or breaching the pool(s). Such an attack is feasible but would require substantial planning and resources and might be repulsed. Thus, this attack option may have a comparatively low probability. It would, however, yield a large release of radioactive material. Option III would be an attack on one or more ISFSI modules using anti-tank missiles fired from one or more offsite locations. In a plausible time window the attackers might, for example, be able to obtain 10 hits. The probability of this option is presumably substantially greater than the probability of Options I and II, but the release of radioactive material would be considerably smaller.

4.3 Consequences of Attack

The offsite radiological consequences of a potential attack on a nuclear facility can be estimated with widely-used, computer-based models. In order to apply such a model, one must have an estimate of the accident "source term". The

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[103] Stone and Webster, 1999.
[104] PG&E, 2001a.

source term is a set of characteristics -- magnitude, timing, etc. -- that describe a potential release of radioactive material to the atmosphere. Using this source term, together with weather data for the release site, the model can estimate the magnitude of each of a range of radiological impacts at specified locations downwind.

Cesium-137 as an Indicator

A full analysis of this type is beyond the scope of this report. Instead, some scoping calculations are presented here, focussing on one radioactive isotope -- cesium-137. This isotope is a useful indicator of the potential, long-term consequences of a release of radioactive material. Cesium-137 has a half-life of 30 years, and accounts for most of the offsite radiation exposure that is attributable to the 1986 Chernobyl reactor accident, and for about half of the radiation exposure that is attributable to fallout from nuclear weapons tests in the atmosphere.[105] Cesium is a volatile element that would be liberally released during nuclear-facility accidents or attacks. For example, an NRC study has concluded that a generic estimate of the release fraction of cesium isotopes during a spent-fuel-pool fire -- that is, the fraction of the pool's inventory of cesium isotopes that would reach the atmosphere -- is 100 percent.[106] It is reasonable to assume such a high release fraction because cesium is volatile, because a fire in a high-density pool, once initiated, would eventually involve all of the fuel in the pool, and because pool buildings are not designed as containment structures.

Inventories of Cesium-137 at Indian Point

The Indian Point site provides an illustration of the inventories of cesium-137 at nuclear facilities. Three nuclear power plants have been built at this site. Unit 1 had a rated power of 590 MW (thermal) and operated from 1962 to 1974.[107] Unit 2 has a rated power of 2,760 MW (thermal), commenced operating in 1974, and remains operational. Unit 3 has a rated power of 2,760 MW (thermal), commenced operating in 1976, and remains operational. Unit 2 and Unit 3 each employ a four-loop Westinghouse PWR with a large, dry containment. The reactor cores of Unit 2 and Unit 3 each contain 193 fuel assemblies.[108]

Unit 2 and Unit 3 are each equipped with one spent-fuel pool. The capacity of the Unit 2 pool is 1,374 fuel assemblies, while the capacity of the Unit 3 pool is

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105 DOE, 1987.
106 Sailor et al, 1987.
107 Thompson and Beckerley, 1973, Table 4-1.
108 Larson, 1985, Table A-2.

1,345 fuel assemblies.[109] Both pools employ high-density racks. As of November 1998, the Unit 2 pool contained 917 fuel assemblies, while the Unit 3 pool contained 672 fuel assemblies.[110] It can be assumed that the number of fuel assemblies in each pool has increased since November 1998.

The inventory of cesium-137 in the Indian Point pools can be readily estimated. Three parameters govern this estimate -- the number of spent fuel assemblies, their respective burnups, and their respective ages after discharge. Assuming a representative, uniform burnup of 46 GW-days per tonne, one finds that the 917 fuel assemblies that were in the Unit 2 pool in November 1998 now contain about 42 million Curies (460 kilograms) of cesium-137. The 672 fuel assemblies that were in the Unit 3 pool in November 1998 now contain about 31 million Curies (350 kilograms) of cesium-137. Additional amounts of cesium-137 would be present in any fuel assemblies that have been added to these pools since November 1998.

For comparison, the cores of the Indian Point Unit 2 and Unit 3 reactors each contain about 6 million Curies (67 kilograms) of cesium-137. Also, it should be noted that the Chernobyl reactor accident of 1986 released about 2.4 million Curies (27 kilograms) of cesium-137 to the atmosphere. That release represented 40 percent of the Chernobyl reactor core's inventory of 6 million Curies (67 kg) of cesium-137.[111] Also, atmospheric testing of nuclear weapons led to the deposition of about 20 million Curies (220 kilograms) of cesium-137 across the land and water surfaces of the Northern Hemisphere.[112]

As another comparison, consider a HI-STORM 100 dry-storage module that contains 32 PWR fuel assemblies. Assuming that these fuel assemblies have an average post-discharge age of 20 years, this module would contain about 1.3 million Curies (14 kilograms) of cesium-137.

Inventories of Cesium-137 at Vermont Yankee

The Vermont Yankee site provides a second illustration of the inventories of cesium-137 at nuclear facilities. At this site there is a single BWR with a rated power of 1,590 MW (thermal) and a Mark I containment. This plant commenced operating in 1972 and remains operational. The reactor core contains 368 fuel assemblies.[113] One spent-fuel pool is provided at this plant.

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109 "Reactor Spent Fuel Storage", from NRC website (www.nrc.gov), 30 May 2001.
110 Ibid.
111 Krass, 1991.
112 DOE, 1987.
113 Larson, 1985, Table A-1.

The pool is equipped with high-density racks and has a capacity of 2,870 fuel assemblies, with a possible recent increase in this capacity.[114]

In 2000, the Vermont Yankee pool contained 2,439 fuel assemblies.[115] Licensee projections done in 1999 showed the pool inventory increasing to a maximum of 2,687 assemblies in 2004, after which the inventory would decrease until the pool would be empty in 2017. These projections assumed continuing operation of the plant until 2012, transfer of spent fuel from the pool to an on-site ISFSI beginning in 2004, and shipment of fuel to Yucca Mountain beginning in 2010.[116] To date, there has been no license application for an ISFSI at Vermont Yankee. Thus, transfer of fuel to an on-site ISFSI in 2004 is unlikely. As discussed in Section 2.1 of this report, shipment of fuel to Yucca Mountain in 2010 is unlikely.

The inventories of cesium-137 in the Vermont Yankee pool and reactor can be estimated as described above for Indian Point. One can assume that the Vermont Yankee pool now (in January 2003) contains 2,639 fuel assemblies, which have been discharged from the reactor during refuelling outages since 1972.[117] Thus, the pool now contains about 35 million Curies (390 kilograms) of cesium-137. The Vermont Yankee reactor contains about 2.3 million Curies (26 kilograms) of cesium-137.

Land Contamination by Cesium-137 After a Pool Fire

Now consider the potential for a spent-fuel-pool fire at Indian Point or Vermont Yankee. As explained above, it is reasonable to assume that 100 percent of the cesium-137 in a pool would be released to the atmosphere in the event of a fire. The cesium-137 would be released to the atmosphere in small particles that would travel downwind and be deposited on the ground and other surfaces. The deposited particles would emit intense gamma radiation, leading to external, whole-body radiation doses to exposed persons. Cesium-137 would also contaminate water and foodstuffs, leading to internal radiation doses.

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114 According to information compiled by licensee staff in February 1999 (Weyman, 1999), the licensed storage limit for the Vermont Yankee pool was 2,870 fuel assemblies in 1999, and was projected to increase to 3,355 fuel assemblies in 2001. According to information compiled by the NRC, the capacity of the Vermont Yankee pool in November 1998 was 2,863 assemblies; see "Reactor Spent Fuel Storage", from NRC website (www.nrc.gov), 30 May 2001.

115 Vermont Yankee, 2000.

116 Weyman, 1999.

117 Ibid.

One measure of the scope of radiation exposure attributable to deposition of cesium-137 is the area of land that would become uninhabitable. For illustration, one can assume that the threshold of uninhabitability is an external, whole-body dose of 10 rem over 30 years. This level of radiation exposure, which would represent about a three-fold increase above the typical level of background (natural) radiation, was used in the NRC's 1975 Reactor Safety Study as a criterion for relocating populations from rural areas.

A radiation dose of 10 rem over 30 years corresponds to an average dose rate of 0.33 rem per year.[118] The health effects of radiation exposure at this dose level have been estimated by the National Research Council's Committee on the Biological Effects of Ionizing Radiations.[119] This committee has estimated that a continuous lifetime exposure of 0.1 rem per year would increase the incidence of fatal cancers in an exposed population by 2.5 percent for males and 3.4 percent for females.[120] Incidence would scale linearly with dose, in this low-dose region.[121] Thus, an average lifetime exposure of 0.33 rem per year would increase the incidence of fatal cancers by about 8 percent for males and 11 percent for females. About 21 percent of males and 18 percent of females normally die of cancer.[122] In other words, in populations residing continuously at the threshold of uninhabitability (an external dose rate of 0.33 rem per year), about 2 percent of people would suffer a fatal cancer that would not otherwise occur.[123] Internal doses from contaminated food and water could cause additional cancer fatalities.

The increased cancer incidence described in the preceding paragraph would apply at the boundary of the uninhabitable area. Within that area, the external dose rate from cesium-137 would exceed the threshold of 10 rem over 30 years. At some locations, the dose rate would exceed this threshold by orders of magnitude. Therefore, persons choosing to live within the uninhabitable area would experience an incidence of fatal cancers at a level higher than is set forth above.

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118 At a given location contaminated by cesium-137, the resulting external, whole-body dose received by a person at that location would decline over time, due to radioactive decay and weathering of the cesium-137. Thus, a person receiving 10 rem over an initial 30-year period would receive a lower dose over the subsequent 30 -year period.

119 National Research Council, 1990.

120 Ibid, Table 4-2.

121 The BEIR V committee assumed a linear dose-response model for cancers other than leukemia, and a model for leukemia that is effectively linear in the low-dose range. See National Research Council, 1990, pp 171-176.

122 National Research Council, 1990, Table 4-2.

123 For males, 0.08 x 0.21 = 0.017. For females, 0.11 x 0.18 = 0.020.

Area of Uninhabitable Land After a Pool Fire at Indian Point or Vermont Yankee

For a postulated release of cesium-137 to the atmosphere, the area of uninhabitable land can be estimated from calculations done by Dr Jan Beyea.[124] Four releases of cesium-137 are postulated here. The first postulated release is 42 million Curies, representing the fuel that was present in the Indian Point Unit 2 pool in November 1998. The second postulated release is 31 million Curies, representing the fuel that was present in the Indian Point Unit 3 pool in November 1998. (Actual, present inventories of cesium-137 in the Unit 2 and Unit 3 pools are higher than these numbers, assuming that fuel has been added since November 1998.) The third postulated release is 35 million Curies, representing the present (January 2003) inventory of fuel in the Vermont Yankee pool. The fourth postulated release is 1 million Curies, representing the cesium-137 inventory in a dry-storage ISFSI module that contains 32 PWR fuel assemblies. This fourth release does not represent a pool fire or a predicted release from an ISFSI. Instead, it is a notional release that provides a scale comparison.

For typical weather conditions, assuming that the radioactive plume travels over land rather than out to sea, a release of 42 million Curies of cesium-137 would render about 95,000 square kilometers of land uninhabitable. Under the same conditions, a release of 31 million Curies would render about 75,000 square kilometers uninhabitable, and a release of 35 million Curies would render about 80,000 square kilometers uninhabitable. A release of 1 million Curies would render uninhabitable about 2,000 square kilometers. For comparison, note that the area of New York state is 127,000 square kilometers, while the combined area of Vermont, New Hampshire and Massachusetts is 70,000 square kilometers. The use of a little imagination shows that a spent-fuel-pool fire at Indian Point or Vermont Yankee would be a regional and national disaster of historic proportions, with health, environmental, economic, social and political dimensions.

Cesium-137 Fallout From a Nuclear Detonation

For attack scenarios involving the use of a nuclear weapon on a spent-fuel-storage facility, it is instructive to compare the long-term radiological significance of the nuclear detonation itself with the significance of the release that the detonation could induce. For example, detonation of a 10-kilotonne fission weapon would directly generate about 2 thousand Curies (21

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124 Beyea et al, 1979.

grams) of cesium-137.[125] Yet, this weapon could release to the atmosphere tens of millions of Curies of cesium-137 from a spent-fuel pool or an unhardened, undispersed ISFSI.

4.4 Defense in Depth

Four types of measure, taken together, could provide a comprehensive, defense-in-depth strategy against acts of malice or insanity at a nuclear facility. The four types of measure, which are described in the following paragraphs, are in the categories: (i) site security; (ii) facility robustness; (iii) damage control; and (iv) emergency response planning. The degree of protection provided by these measures would be greatest if they were integrated into the design of a facility before its construction. However, a comprehensive set of measures could provide significant protection at existing facilities.

Site Security

Site-security measures are those that reduce the potential for implementation of destructive acts of malice or insanity at a nuclear site. Two types of measure fall into this category. Measures of the first type would be implemented at offsite locations, and the implementing agencies might have no direct connection with the site. Airline or airport security measures are examples of measures in this category. Measures of the second type would be implemented at or near the site. Implementing agencies would include the licensee, the NRC and, potentially, other entities (e.g., National Guard, Coast Guard). The physical protection measures now required by the NRC, as discussed in Section 2.3 of this report, are examples of site-security measures of the second type. More stringent measures could be introduced, such as:

(i) establishment of a mandatory aircraft exclusion boundary around the site;
(ii) deployment of an approaching-aircraft detection system that triggers a high-alert status at facilities on the site;
(iii) expansion of the DBT, beyond that now applicable to a nuclear power plant, to include additional intruders, heavy weapons, lethal chemical weapons and more than one vehicle bomb; and
(iv) any ISFSI on the site to receive protection equivalent to that provided for a nuclear power plant.

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[125] SIPRI, 1981, page 76.

Facility Robustness

Facility-robustness measures are those that improve the ability of a nuclear facility to experience destructive acts of malice or insanity without a significant release of radioactive material to the environment. In illustration, the PIUS reactor design, as discussed in Section 2.3, was intended to withstand aerial bombardment by 1,000-pound bombs without suffering core damage or releasing a significant amount of radioactive material to the environment. An ISFSI could be constructed with a similar degree of robustness. At existing facilities, a variety of opportunities are available for enhancing robustness. As a high-priority example, the spent fuel pool(s) at a nuclear power plant could be re-equipped with low-density racks, so that spent fuel would not ignite if water were lost from a pool. As a second example, the reactor of a nuclear power plant could be permanently shut down, or the reactor could operate at reduced power, either permanently or at times of alert. Other robustness-enhancing opportunities could be identified. For a nuclear power plant whose reactor is not permanently shut down, robustness could be enhanced by an integrated set of measures such as:

(i) automated shutdown of the reactor upon initiation of a high-alert status at the plant, with provision for completion of the automated shutdown sequence if the control room is disabled;
(ii) permanent deployment of diesel-driven pumps and pre-engineered piping to be available to provide emergency water supply to the reactor, the steam generators (at a PWR) and the spent fuel pool(s);
(iii) re-equipment of the spent fuel pool(s) with low-density racks, excess fuel being stored in an onsite ISFSI; and
(iv) construction of the ISFSI to employ hardened, dispersed, dry storage.

Damage Control

Damage-control measures are those that reduce the potential for a release of radioactive material from a facility following damage to that facility due to destructive acts of malice or insanity. Measures of this kind could be ad hoc or pre-engineered. One illustration of a damage control measure would be a set of arrangements for patching and restoring water to a spent fuel pool that has been breached. Many other illustrations can be provided. It appears, from the list of additional measures set forth in Section 2.3 of this report, that the NRC's recent orders have required licensees to undertake some planning for damage control following explosions or fires. Additional measures would be appropriate. For example, at a site housing one or more nuclear power plants and an ISFSI, the following damage-control measures could be implemented:

(i) establishment of a damage control capability at the site, using onsite personnel and equipment for first response and offsite resources for backup;
(ii) periodic exercises of damage-control capability;
(iii) establishment of a set of damage-control objectives -- to include patching and restoring water to a breached spent fuel pool, fire suppression in the ISFSI, and provision of cooling to a reactor whose support systems and control room are disabled -- with accompanying plans; and
(iv) provision of equipment and training to allow damage control to proceed on a radioactively-contaminated site.

Offsite Emergency Response

Emergency-response measures are those that reduce the potential for exposure of offsite populations to radiation, following a malice- or insanity-induced release of radioactive material from a nuclear facility. Measures in this category would in many respects be similar to emergency planning measures that are designed to accommodate "accidental" releases of radioactive material arising from human error, equipment failure or natural forces (e.g., earthquake). However, there are two major ways in which malice- or insanity-induced releases might differ from accidental releases. First, a malice- or insanity-induced release might be larger and begin earlier than an accidental release.[126] Second, a malice- or insanity-induced release might be accompanied by deliberate degradation of emergency response capabilities (e.g., the attacking group might block an evacuation route). Accommodating these differences could require additional measures of emergency response. Overall, an appropriate way to improve emergency-response capability at a nuclear-power-plant site could be to implement a model emergency response plan that was developed by a team based at Clark University in Massachusetts.[127] This model plan was specifically designed to accommodate radioactive releases from spent-fuel-storage facilities, as well as from reactors. That provision, and other features of the plan, would provide a capability to accommodate both accidental releases and malice- or insanity-induced releases. Major features of the model plan include:[128]

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126 Present plans for emergency response do not account for the potential for a large release of radioactive material from spent fuel, as would occur during a pool fire. The underlying assumption is that a release of this kind is very unlikely. That assumption cannot be sustained in the present threat environment.
127 Golding et al, 1992.
128 Ibid, pp 8-13.

(i) structured objectives;
(ii) improved flexibility and resilience, with a richer flow of information;
(iii) precautionary initiation of response, with State authorities having an independent capability to identify conditions calling for a precautionary response[129];
(iv) criteria for long-term protective actions;
(v) three planning zones, with the outer zone extending to any distance necessary[130];
(vi) improved structure for accident classification;
(vii) increased State capabilities and power;
(viii) enhanced role for local governments;
(ix) improved capabilities for radiation monitoring, plume tracking and dose projection;
(x) improved medical response;
(xi) enhanced capability for information exchange;
(xii) more emphasis on drills, exercises and training;
(xiii) improved public education and involvement; and
(xiv) requirement that emergency preparedness be regarded as a safety system equivalent to in-plant systems.

4.5 A Strategy for Robust Storage of Spent Fuel

The preceding section of this report sets forth a defense-in-depth strategy for nuclear facilities. This strategy could be implemented at every civilian nuclear facility in the United States. Within the context of that strategy, it would be necessary to establish a nationwide strategy for the robust storage of spent fuel. The strategy must protect all spent fuel that has been discharged from a reactor but has not been emplaced in a repository. Available options for storing this fuel are wet storage in pools and dry storage in ISFSIs.

Timeframe for a Robust-Storage Strategy

As pointed out in Section 2.1 of this report, thousands of tonnes of US spent fuel will remain in interim storage for decades, even if a repository opens at Yucca Mountain. If a repository does not open, the entire national inventory of spent fuel will remain in interim storage for many decades. Thus, the robust-storage strategy for spent fuel must minimize the overall risk of interim storage throughout a period that may extend for 100 years or longer.

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129 A security alert could be a condition calling for a precautionary response.

130 The inner and intermediate zones would have radii of 5 and 25 miles, respectively. As an example of the planning measures in each zone, potassium iodide would be predistributed within the 25-mile zone and made generally accessible nationwide.

Moreover, this interim storage strategy must be compatible with the eventual emplacement of the spent fuel in a repository in a manner that minimizes long-term risk.

Reactor Risk and Spent-Fuel Risk

This report focusses on the risk of a radioactive release from spent fuel. It also, by necessity, discusses the risk of a similar release from a reactor. These risks are closely intertwined in two practical ways. First, many scenarios for a spent-fuel-pool fire involve interactions between the affected pool(s) and the reactor(s) on the site. Second, the security of an at-reactor ISFSI is an adjunct to the security of a nuclear-power-plant site.

A robust-storage strategy for spent fuel could substantially reduce the risk of a radioactive release from spent fuel, at a comparatively low cost. Given the design of US nuclear power plants, there is no obvious strategy for achieving a comparable reduction in reactor risk. Thus, even if a defense-in-depth strategy is implemented for every reactor, a substantial fraction of the present reactor risk will continue to exist as long as the reactors continue to operate.

What should be the risk target for a robust-storage strategy? There are three major considerations that argue for seeking a spent-fuel risk that is substantially lower than the reactor risk. First, measures are available for substantially reducing the spent-fuel risk at a comparatively low cost. Second, storing spent fuel creates no benefit to offset its risk, whereas reactors generate electricity. Third, spent fuel may be in interim storage for 100 years or longer, whereas the present reactors will operate for at most a few more decades.

Elements of a Robust-Storage Strategy

From Sections 4.2 and 4.3 of this report, it is evident that storing spent fuel in high-density pools poses a very high risk. Dry storage of spent fuel, even employing the present practice that is described in Section 2.3, poses a lower risk. Thus, a robust-storage strategy must assign its highest priority to re-equipping each spent fuel pool with low-density racks, in order to reduce the pool's inventory of fuel and to prevent self-ignition and burning of fuel if water is lost from the pool.[131] The excess fuel, for which space would no longer be available in pools, would be transferred to ISFSIs. When a nuclear power plant is shut down, the fuel remaining in its pool(s) would be transferred to an ISFSI after an appropriate period of cooling. These steps would dramatically reduce the overall risk of spent-fuel storage. A further,

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131 Further protection of the spent fuel that remains in pools could be provided by a variety of site-security, facility-robustness and damage-control measures of the kind that are described in Section 4.4 of this report.

substantial reduction of the overall risk would be obtained by employing hardened, dispersed, dry storage at every ISFSI.

Figure 1, on the following page, shows how a robust-storage strategy for spent fuel would operate in a larger context. The robust-storage strategy would have the three elements represented by the three boxes at the base of the figure: low-density pools; hardened dry-storage modules; and dispersed dry-storage modules. In turn, the robust-storage strategy would be one of the elements of facility robustness, which itself would be one of four components of a defense in depth for US civilian nuclear facilities. This defense would contribute to homeland security and national security.

Away-from-Reactor ISFSIs

In a robust-storage strategy, any ISFSI would employ hardened, dispersed dry storage. The essential principles would be the same whether the ISFSI is at a nuclear-power-plant site or at another site such as Skull Valley.

Section 2.1 of this report discusses factors that argue against shipping spent fuel to an away-from-reactor ISFSI. Some of these factors are economic in nature. However, three factors affect the overall risk of interim storage. First, shipment to an away-from-reactor ISFSI would increase the overall transport risk, because fuel would be shipped twice, first from the reactor site to the ISFSI, and then from the ISFSI to the ultimate repository. Second, an away-from-reactor ISFSI would hold a comparatively large inventory of spent fuel, creating a potentially attractive target for an enemy. Third, there is a risk that a large, away-from-reactor ISFSI would become, by default, a permanent repository, despite having no long-term containment capability. These three factors must be considered in minimizing the overall risk of interim storage.

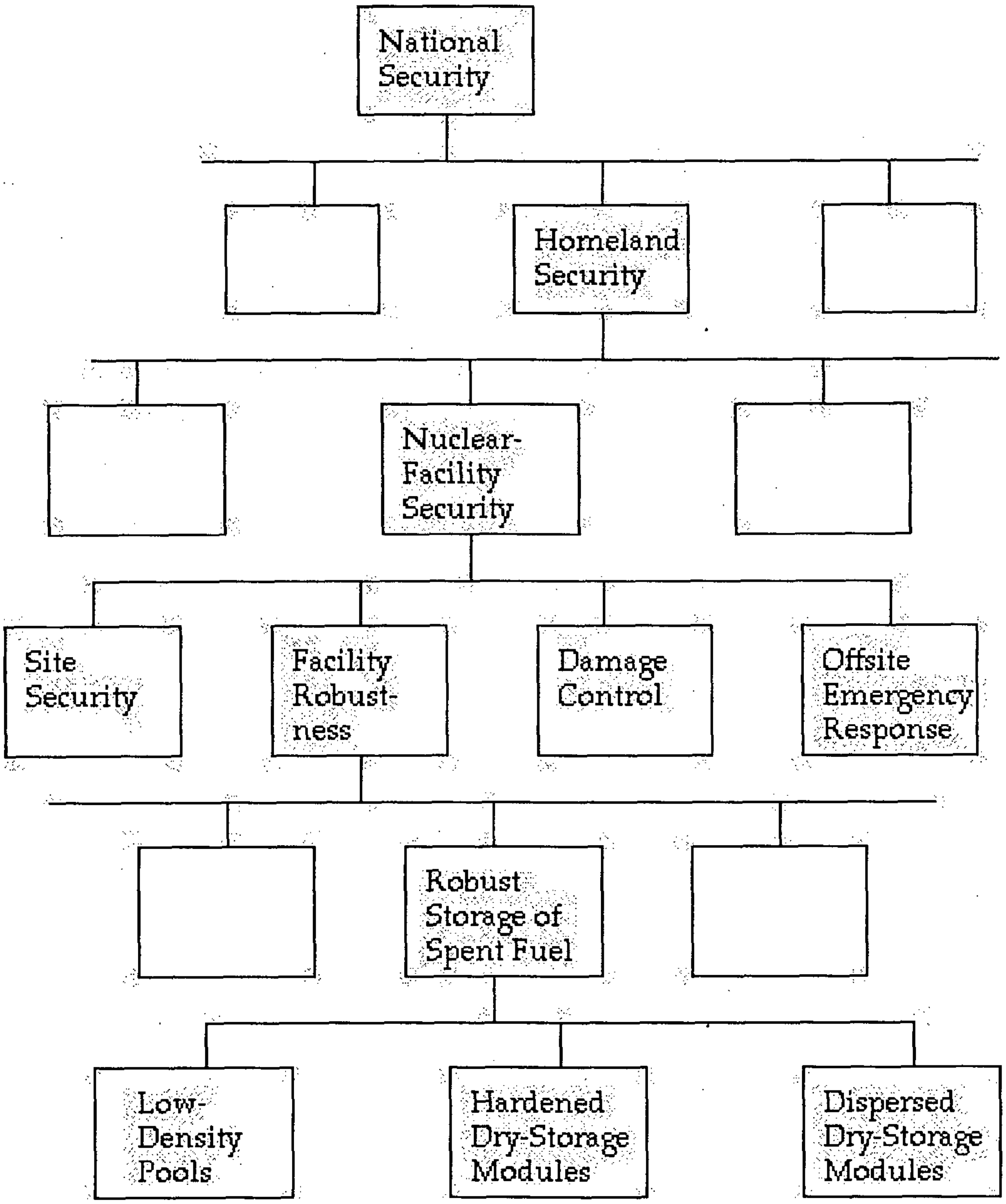

FIGURE 1

ROBUST STORAGE OF SPENT FUEL
IN THE CONTEXT OF NATIONAL SECURITY

5. Considerations in Planning Hardened, Dispersed, Dry Storage

5.1 Balancing Short- and Long-Term Risks

Interim storage of spent fuel could lead to eventual emplacement of the fuel in a repository at Yucca Mountain. In this case, fuel would remain in interim storage for several decades. That period is long enough to require action to reduce the very high risk that is posed by pool storage, and the smaller but still significant risk that is posed by unhardened, undispersed ISFSIs. However, in this case the long-term risk posed by spent-fuel management would not be relevant to interim storage. The long-term risk, which will be significant for many thousands of years, would be associated with the Yucca Mountain repository.

Avoiding a Repository by Default

If a repository does not open, a different problem will arise. That problem is the possibility that society will extend the life of interim-storage facilities until they become, by default, repositories for spent fuel. These facilities would function poorly as repositories, and the environment around each facility would become contaminated by radioactive material leaking from the facility. This outcome would pose a substantial long-term risk. The prospect of society acting in this improvident manner may seem far-fetched, but becomes more credible when one examines the history of the Yucca Mountain project. That project is politically driven, and is going forward only because previously-specified technical criteria for a repository have been abandoned.[132]

Any current planning for the implementation of interim storage must account for the possibility that a repository will not open at Yucca Mountain. Thus, the design approach that is adopted for a hardened, dispersed, dry-storage ISFSI must balance two objectives. <u>The first objective</u> is that the facility should be comparatively robust against attack. <u>The second objective</u> is that the facility should not have features that encourage society to allow the facility to become, by default, a repository.

Consideration of the second objective dictates that the ISFSI should not, unless absolutely necessary, be located underground. Therefore, the first objective should be pursued through a design in which the ISFSI modules are stored at grade level (i.e., at the general level of the site). Hardening would then be achieved by placing steel, concrete, gravel or other materials above

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[132] Ewing and Macfarlane, 2002.

and around each module. The remaining protection would be provided by dispersal of the storage modules.

5.2 Cost and Timeframe for Implementation

As discussed in Section 2.1 of this report, forecasts show a rapid expansion in dry-storage capacity across the USA over the coming years. NAC International predicts that about 30 percent of US commercial spent fuel will be in dry-storage ISFSIs by 2010, as compared with 6 percent at the end of 2000. Vendors have developed a comparatively cheap technology for these ISFSIs, in response to to industry preferences. This technology -- the overpack system -- involves the placement of spent fuel into thin-walled metal containers that are stored inside overpacks made primarily from concrete. The resulting modules are placed close together in large numbers on concrete pads in the open air. A preference for vertical-axis modules seems to be emerging.

Required Properties of Dry-Storage Modules

Re-equipping US spent fuel pools with low-density racks would create a large additional demand for dry-storage modules. This demand should be met as quickly as possible, in view of the very high risk that is posed by high-density pool storage. Also, the cost of the additional storage capacity should be minimized, consistent with the achievement of performance objectives. Thus, it is desirable that module designs already approved by the NRC be used. However, any module that is used for a hardened, dispersed ISFSI must be capable, when hardened, of resisting a specified attack. This requirement did not exist when module designs were approved by the NRC. Also, it is desirable that modules be capable of retaining their integrity for 100 years or more, which was not a requirement when module designs were approved by the NRC. A module that does not have a long-life capability may need to be replaced at some point if it is used in an ISFSI that serves for an extended period. Finally, the design of a module should allow for the eventual transport of spent fuel from an ISFSI to a repository.

Meeting the Requirements:
Monolithic Casks versus Overpack Systems

Of the module designs already approved by the NRC, monolithic casks such as the CASTOR are probably more capable of meeting attack-resistance and long-life requirements than are modules that employ a thin-walled metal container inside a concrete overpack. However, monolithic casks are more expensive. Thus, it would be convenient if some of the cheaper and more widely-used module designs proved to be capable of meeting attack-resistance and long-life requirements. This outcome would minimize the cost of offloading fuel from pools to hardened, dispersed dry storage, and would expedite this transition.

The development of detailed requirements for attack resistance and long life is a task beyond the scope of this report. Section 7 of the report sets forth a process for developing attack-resistance requirements, drawing upon experiments. When that process is completed, it will be possible to determine which of the already-approved module designs can be used for hardened, dispersed, dry storage.

5.3 Design-Basis Threat

The specification of a DBT for a nuclear facility inevitably reflects a set of tradeoffs. In the case of a hardened, dispersed, dry-storage ISFSI, five major considerations must be balanced. First, the ISFSI must protect spent fuel against a range of possible attacks. Second, the cost of the ISFSI should not be dramatically higher than the cost of an ISFSI built according to present practice. Third, the timeframe for building of the ISFSI should be similar to the timeframe for building an ISFSI according to present practice. Fourth, the ISFSI should not, unless absolutely necessary, be built underground. Fifth, it should be possible to construct an ISFSI of this kind at every US nuclear-power-plant site.

These considerations suggest a two-tier DBT for a hardened, dispersed, dry-storage ISFSI. This DBT might have the following structure:

<u>Tier I</u>

There should be high confidence that the release of radioactive material from the ISFSI to the environment would not exceed a small, specified amount in the event of a direct attack on any part of the ISFSI by:

(i) a TOW missile;
(ii) a specified manually-placed charge;
(iii) a specified vehicle bomb;
(iv) a specified explosive-laden general-aviation aircraft; or
(v) a fuel-laden commercial aircraft.

<u>Tier II</u>

There should be reasonable confidence that the release of radioactive material from the ISFSI to the environment would not exceed a specified amount in the event of a ground burst, at any part of the ISFSI, of a 10-kilotonne nuclear weapon.

5.4 Site Constraints

At each ISFSI site there will be a site-specific set of constraints on the development of a hardened, dispersed ISFSI. Some constraints will be political, financial or in some other non-physical category. Other constraints will be physical, reflecting the geography of the site. Of the physical constraints, the most significant will be the land area required for dispersal of dry-storage modules.

At many nuclear-power-plant sites, ample land area will be available for dispersal. At some, smaller sites, it may not be possible to achieve the desired degree of dispersal, but this deficiency might be compensated by increased hardening. At the smallest sites, it might be necessary to relax the requirement that the ISFSI should not be built underground. This step would allow a substantial increase in hardening, to offset the limited degree of dispersal that could be achieved. At especially-constricted sites, it might be necessary to ship some spent fuel from the site to an ISFSI elsewhere.

6. A Proposed Design Approach for Hardened, Dispersed, Dry Storage

An ISFSI design approach that offers a prospect of meeting the above-specified DBT involves an array of vertical-axis dry-storage modules at a center-to-center spacing of perhaps 25 meters. Each module would be on a concrete pad slightly above ground level, and would be surrounded by a concentric tube surmounted by a cap, both being made of steel and concrete. This tube would be backed up by a conical mound made of earth, gravel and rocks. Further structural support would be provided by triangular panels within the mound, buttressing the tube. The various structural components would be tied together with steel rods. Air channels would be provided, to allow cooling of the dry-storage module. These channels would be inclined, to prevent pooling of jet fuel, and would be configured to preclude line-of-sight access to the dry-storage module. Figure 2, on the following page, provides a schematic view of the proposed design.

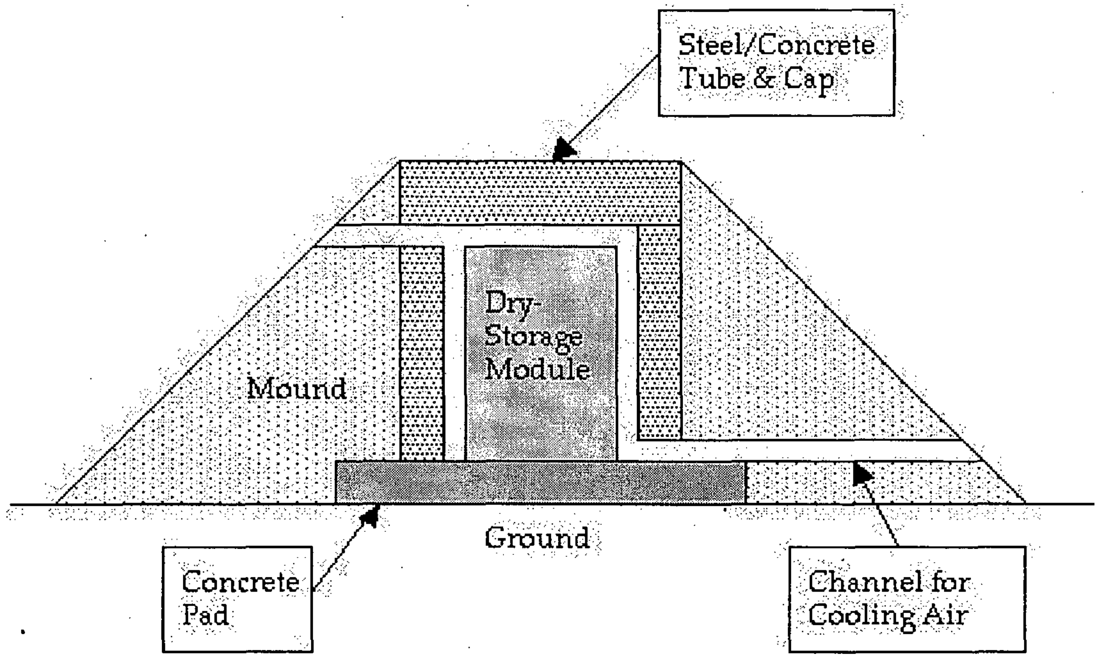

FIGURE 2

SCHEMATIC VIEW OF PROPOSED DESIGN FOR HARDENED, DRY STORAGE

Notes

1. Cooling channels would be inclined, to prevent pooling of jet fuel, and would be configured to preclude line-of-sight access to the dry-storage module.
2. The tube, cap and pad surrounding the dry-storage module would be tied together with steel rods, and spacer blocks would prevent the module from moving inside the tube.
3. The steel/concrete tube could be buttressed by several triangular panels connecting the tube and the base pad.

Further analysis and full-scale experiments would be needed to determine whether this design approach, or something like it, could meet the DBT and other requirements that are set forth in Section 5, above. Ideally, these requirements could be met while using dry-storage modules that are approved by the NRC and are in common use. Another objective would be that the hardening elements (concentric tube, cap, tie rods, mound, etc.) could be built and assembled comparatively quickly and cheaply. These elements would not be high-technology items.

The Benefits of Dispersal

As an illustration of the benefits of dispersal, consider an attack on an ISFSI involving a ground burst of a 10-kilotonne nuclear weapon. In Section 4.2 of this report, it was noted that this attack could excavate a crater about 68 meters in diameter and 16 meters deep. If dry-storage modules had a center-to-center spacing of 5.5 meters, as is typical of present practice, about 120 modules could fall within the crater area and suffer destruction. However, if the center-to-center spacing were 25 meters, as is proposed here, only 6 modules could fall within the crater area and suffer destruction.

Site-Specific Tradeoffs

Within this design approach it would be possible to trade off, to some extent, hardening and dispersal. As suggested in Section 5.4, above, dispersal could be reduced and hardening could be increased at smaller sites. Detailed, site-specific analysis is needed to determine how such tradeoffs might work.

An alternative design approach might be used at a few sites where space is insufficient to allow wide dispersal. In this approach, a number of dry-storage modules would be co-located in an underground, reinforced-concrete bunker. Similar bunkers would be dispersed across the site to the extent allowed by the site's geography. At an especially-constricted site, it might be necessary to reduce the overall inventory of spent fuel in order to meet design objectives. Thus, some spent fuel from the site would be shipped to an ISFSI elsewhere.

7. Requirements for Nationwide Implementation of Robust Storage

7.1 Experiments on Vulnerability of Dry-Storage Options

Section 5.3 of this report outlines a DBT for hardened, dispersed, dry storage of spent fuel. Section 6 describes a design approach that offers a prospect of meeting a DBT of this kind, together with other requirements that are set forth in Section 5. Further investigation is needed to determine the extent to which the various requirements can be met. This determination would be made at two levels. First, the investigation would determine if the DBT and other requirements set forth in Section 5 are broadly compatible with the proposed design approach or something like it. Second, assuming an affirmative determination at the first level, the investigation would go into more detail, exploring the various tradeoffs that could be made.

An essential part of this investigation would be a series of full-scale, open-air experiments. These experiments would be sponsored by the US government, and would be conducted at US government laboratories and testing centers. The experiments would involve a range of non-nuclear instruments of attack, including anti-tank missiles, manually-placed charges, vehicle bombs and aircraft bombs. Each instrument of attack would be tested against several test specimens that would simulate alternative design approaches for a hardened, dispersed ISFSI.

A separate set of experiments would be conducted in contained situations. These experiments would study the potential for release of radioactive material following penetration or prolonged heating of a fuel container.[133] Factors discussed in Section 4.2 of this report, such as the presence of zirconium hydride in fuel cladding, would be accounted for. The potential for auto-ignition of hydrided cladding when exposed to air deserves special attention in the experimental program, because this potential is relevant not only to the vulnerability of dry-storage modules, but also to the initiation of a fire in a spent-fuel pool.[134]

7.2 Performance-Based Specifications for Robust Storage

The investigation called for in Section 7.1 would establish the technical basis for a set of performance-based specifications for hardened, dispersed, dry storage of spent fuel. These specifications would include a detailed, precise formulation of the DBT. Also included would be design guidelines for meeting the DBT, and an allowable range of design parameters within which tradeoffs could be made. The specifications would apply not only to the design of external, hardening elements, but also to dry-storage modules. Thus, some modification of the licensing basis for currently-licensed dry-storage modules may be required.

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[133] The proposed experiments would simulate, among other events, an attack in which penetration of a fuel container is accompanied by incendiary effects.

[134] At the higher fuel burnups now commonly achieved, zirconium hydride forms in the fuel cladding. A potential for auto-ignition of zirconium hydride in air has been identified. See: Powers, 2000, page 3; Collins et al, 2000, page A1B-3.

Specifications for Low-Density Pool Storage

Performance specifications would also be required for the nationwide reversion to low-density pool storage. A primary objective would be to prevent the initiation of a pool fire in the event of a loss of water from a pool. This would be accomplished by reverting to low-density, open-frame racks that allow convective cooling of fuel by air or steam in the event of water loss, as discussed in Section 4.2. (Note: Low-density, open-frame racks would not necessarily preclude a pool fire after water loss if auto-ignition of zirconium hydride, as discussed in Section 7.1, could occur. Thus, it is important to empirically resolve the auto-ignition issue.)

At nuclear power plants with larger pools, reverting to low-density, open-frame racks will not conflict with other objectives. At plants with smaller pools, the pursuit of low density may conflict with other objectives, including: (i) preserving open spaces in the racks to allow offloading of the reactor core; (ii) allowing fuel to age for at least 5 years before transferring it to an ISFSI; and (iii) suppressing criticality of fresh or low-burnup fuel without relying on soluble boron in the pool water. Tradeoffs and technical fixes could resolve many of these conflicts.[135] New analysis, perhaps supplemented by some experiments, would establish the technical basis for performance specifications that include the necessary tradeoffs.

Establishing the Specifications

Establishing a comprehensive set of specifications for robust storage would call for the exercise of judgement. There is no purely objective basis for deciding upon one level of required performance as opposed to another. However, judgement must be exercised with full awareness of the wide-ranging implications of a particular choice. As discussed in Section 3 of this report, the defense of US nuclear facilities should be seen as a key component of homeland security and international security.

In view of the national importance of the needed set of specifications, these should be developed with the full engagement of stakeholders. Relevant stakeholders include citizen groups, local governments and state

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135 Examples of possible tradeoffs and technical fixes include: (i) relaxing the requirement to offload a full core; (ii) providing some high-density rack spaces for fresh fuel and core offload; (iii) relying on soluble boron in normal operation, with limited addition of unborated water if borated water is lost; (iv) adding some solid boron to rack structures while preserving an open-frame configuration; (v) relaxing the 5-year cooling period by partially filling some dry-storage modules or mixing younger fuel with older fuel in dry-storage modules; and (vi) shipping some fuel to plants with larger pools.

governments. Processes are available that could allow full engagement of stakeholders while protecting sensitive information.[136]

7.3 A Homeland-Security Strategy for Robust Storage

A robust-storage strategy for US spent fuel would involve two major initiatives. The first initiative would be to re-equip the nation's spent-fuel pools with low-density racks and to provide other defense-in-depth measures to protect the pools. The second initiative would be to place all spent fuel, other than the residual amount that would then be stored in low-density pools, into hardened, dispersed, dry-storage ISFSIs.

Fast, effective implementation of this strategy would require decisive action by the US government. It would require expenditures that are comparatively small by national-security standards but are nonetheless significant. At present, there is no sign that the needed action will be taken. The US government in general seems largely unaware of the threat posed by the present practice of storing spent fuel. The NRC appears to be paralyzed, perhaps through fear of being criticized for its previous inattention to the threat of attack on nuclear facilities.

A new paradigm is needed, in which spent-fuel-storage facilities are seen as pre-deployed radiological weapons that await activation by an enemy. Correcting this situation is an imperative of national defense. If the NRC continues to undermine national defense, it should be bypassed. Citizens should insist that Congress and the executive branch promptly initiate a strategy for robust storage of spent fuel, as a key element of homeland security.

8. Conclusions

The prevailing practice of storing most US spent fuel in high-density pools poses a very high risk because knowledgeable attackers could induce a loss of water from a pool, causing a spent-fuel fire that would release a huge amount of radioactive material to the atmosphere. Nuclear reactors are also vulnerable to attack. Dry-storage modules used in ISFSIs have safety advantages in comparison to pools and reactors, but are not designed to resist a determined attack.

Thus, nuclear power plants and their spent fuel can be regarded as pre-deployed radiological weapons that await activation by an enemy. The US government in general and the NRC in particular seem unaware of this

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[136] Thompson, 2002a, Sections IX and X.

threat. US nuclear facilities are lightly defended and are not designed to resist attack. This situation is symptomatic of an unbalanced US strategy for national security, which is a potentially destabilizing factor internationally.

A strategy for robust storage of US spent fuel is needed, whether or not a repository is opened at Yucca Mountain. This strategy should be implemented as a major element of a defense-in-depth strategy for US civilian nuclear facilities. In turn, that defense-in-depth strategy should be a component of a homeland-security strategy that provides solid protection of our critical infrastructure.

The highest priority in a robust-storage strategy for spent fuel would be to re-equip spent-fuel pools with low-density, open-frame racks. As a further measure of risk reduction, ISFSIs should be re-designed to incorporate hardening and dispersal. These measures should not be implemented in a manner such that an ISFSI may become, by default, a repository. Therefore, a hardened ISFSI should not, unless absolutely necessary, be built underground. Also, the cost and timeframe for implementing hardening and dispersal should be minimized. These considerations argue for the use, if possible, of dry-storage modules that are already approved by the NRC and are in common use.

Preliminary analysis suggests that a hardened, dispersed ISFSI meeting these criteria could be designed to meet a two-tiered DBT. The first tier would require high confidence that no more than a small release of radioactive material would occur in the event of a direct attack on the ISFSI by various non-nuclear instruments. The second tier would require reasonable confidence that no more than a specified release of radioactive material would occur in the event of attack using a 10-kilotonne nuclear weapon.

Three major requirements must be met if a robust-storage strategy for spent fuel is to be implemented nationwide. First, appropriate experiments are needed. Second, performance-based specifications for robust storage must be developed with stakeholder involvement. Third, robust storage for spent fuel must be seen as a vital component of homeland security.

9. Bibliography

(Alpert et al, 1986)
Daniel J. Alpert et al, Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions, NUREG/CR-4467 (Washington, DC: NRC, March 1986).

(ASLB, 1982)
Atomic Safety and Licensing Board, "Carolina Power and Light Co." (Shearon Harris Nuclear Power Plant, Units 1 and 2), LBP-82-119A, 16 NRC 2069 (1982).

(Baker, Cutler et al, 2001)
Howard Baker and Lloyd Cutler (co-chairs, Russia Task Force) et al, A Report Card on the Department of Energy's Nonproliferation Programs with Russia (Washington, DC: Secretary of Energy Advisory Board, US Department of Energy, 10 January 2001).

(Barnard and Kerber, 2001)
Anne Barnard and Ross Kerber, "Web posting tells of suspect's firing from Maine plant", The Boston Globe, 5 January 2001, page A12.

(Betts, 2003)
Richard K. Betts, "Suicide From Fear of Death", Foreign Affairs, January/February 2003, pp 34-43.

(Beyea et al, 1979)
Jan Beyea, Yves Lenoir, Gene Rochlin and Gordon Thompson (subgroup chair), Report of the Gorleben International Review, Chapter 3: Potential Accidents and their Effects, submitted (in German) to the Government of Lower Saxony, March 1979.

(Bunn et al, 2002)
Matthew Bunn, John P. Holdren and Anthony Wier, Securing Nuclear Weapons and Materials: Seven Steps for Immediate Action (Cambridge, Massachusetts: John F. Kennedy School of Government, Harvard University, May 2002).

(Burns, 2002)
Jimmy Burns, "Al-Qaeda planning attack on US interests", Financial Times, 5 September 2002.

(Carter et al, 1998)
A. Carter, J. Deutch and P. Zelikow, "Catastrophic Terrorism", Foreign Affairs, November/December 1998.

(Chapin et al, 2002)
Douglas M. Chapin et al, "Nuclear Power Plants and Their Fuel as Terrorist Targets", Science, Volume 297, 20 September 2002, pp 1997-1999.

(Collins et al, 2000)
Timothy Collins et al, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (Washington, DC: US Nuclear Regulatory Commission, October 2000).

(Commission on National Security, 1999)
US Commission on National Security/21st Century, New World Coming: American Security in the 21st Century, Phase I report, 15 September 1999.

(Commons Defence Committee, 2002)
House of Commons Defence Committee, Defence and Security in the UK: Sixth Report of Session 2001-02 (London: The Stationery Office Limited, 24 July 2002).

(Conetta, 2002)
Carl Conetta, Dislocating Alcyoneus: How to combat al-Qaeda and the new terrorism (Cambridge, Massachusetts: Project on Defense Alternatives, 25 June 2002).

(Crawford, 2003)
Alan Crawford, "Terrorism 'not as bad as war': Leading academic warns 'disproportionate response' by Britain and US forces could escalate conflict", Sunday Herald, 12 January 2003.

(Decker, 2001)
Raymond J. Decker, Director, Defense Capabilities and Management, US General Accounting Office, "Homeland Security: A Risk Management Approach Can Guide Preparedness Efforts", testimony before the Senate Committee on Governmental Affairs, 31 October 2001.

(Deller, 2002)
Nicole Deller, "Rule of Power or Rule of Law?", Science for Democratic Action, Volume 10, Number 4, August 2002.

(DOE, 2002)
US Department of Energy, Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada, DOE/EIS-0250F (Washington, DC: DOE, February 2002).

(DOE, 1987)
US Department of Energy, Health and Environmental Consequences of the Chernobyl Nuclear Power Plant Accident, DOE/ER-0332 (Washington, DC: DOE, June 1987).

(ERIN, 2000)
ERIN Engineering and Research Inc, "Technical Input for Use in the Matter of Shearon Harris Spent Fuel Pool Before the Atomic Safety and Licensing Board (Docket No. 50-400-LA)", November 2000.

(Ewing and Macfarlane, 2002)
Rodney C. Ewing and Allison Macfarlane, "Yucca Mountain", Science, Volume 296, 26 April 2002, pp 659-660.

(Farah, 2002)
Douglas Farah, "Al Qaeda, Taliban said to stash gold in Sudan", The Boston Globe, 3 September 2002, page A11.

(Fetter, 1982)
Steve Fetter, The Vulnerability of Nuclear Reactors to Attack by Nuclear Weapons (Cambridge, Massachusetts: Physics Department, Massachusetts Institute of Technology, August 1982).

(Fetter and Tsipis, 1980)
Steve Fetter and Kosta Tsipis, Catastrophic Nuclear Radiation Releases (Cambridge, Massachusetts: Physics Department, Massachusetts Institute of Technology, September 1980).

(Flowers et al, 1976)
Brian Flowers (chair) et al, Nuclear Power and the Environment: Sixth Report of the Royal Commission on Environmental Pollution, Cmnd. 6618 (London: HMSO, September 1976).

(Garwin, 2001)
Richard Garwin, "The Many Threats of Terror", The New York Review, 1 November 2001.

(Gervasi, 1977)
Tom Gervasi, Arsenal of Democracy (New York: Grove Press, 1977).

(Glasstone, 1962)
Samuel Glasstone (editor), The Effects of Nuclear Weapons (Washington, DC: US Atomic Energy Commission, April 1962).

(Golding et al, 1995)
Dominic Golding et al, Preparing for Nuclear Power Plant Accidents (Boulder, Colorado: Westview Press, 1995).

(Golding et al, 1992)
Dominic Golding et al, Managing Nuclear Accidents: A Model Emergency Response Plan for Power Plants and Communities (Boulder, Colorado: Westview Press, 1992).

(Goldman, 2000)
John J. Goldman, "Former sergeant admits role in bombings of US embassies", The Boston Globe, 21 October 2000, page A2.

(Hannerz, 1983)
K. Hannerz, Towards Intrinsically Safe Light Water Reactors (Oak Ridge, Tennessee: Institute for Energy Analysis, February 1983).

(Hart et al, 2002)
Gary Hart and Warren B. Rudman (co-chairs), Stephen E. Flynn (project director) and Task Force members, America Still Unprepared -- America Still in Danger: Report of an Independent Task Force Sponsored by the Council on Foreign Relations (New York, NY: Council on Foreign Relations, 25 October 2002).

(von Hippel, 2001)
Frank von Hippel, "Recommendations for Preventing Nuclear Terrorism", FAS Public Interest Report: Journal of the Federation of American Scientists, Volume 54, Number 6, November/December 2001, pp 1-10.

(Hirsch, 2002)
Helmut Hirsch, unpublished note on the safety and security of German dry-storage facilities that use CASTOR casks, summarizing the findings of German-language reports that have been prepared by Hirsch and others for the Austrian Federal Government and Austrian State Governments, September 2002.

(Hirsch et al, 1989)
H. Hirsch et al, IAEA Safety Targets and Probabilistic Risk Assessment (Hannover, Germany: Gesellschaft fur Okologische Forschung und Beratung mbH, August 1989).

(Hoffman, 2002)
F. G. Hoffman, Homeland Security: A Competitive Strategies Approach (Washington, DC: Center for Defense Information, March 2002).

(Hogg, 1991)
Ian V. Hogg (editor), Jane's Infantry Weapons, 1991-92 (Alexandia, Virginia: Jane's Information Group, 1991)

(Jackson, 1996)
Paul Jackson (editor), Jane's All the World's Aircraft, 1996-97 (Alexandria, Virginia: Jane's Information Group, 1996).

(Janberg, 2002)
Klaus Janberg, oral presentation at the MIT seminar, "Radiological Hazard Posed by Nuclear Power Plants and Spent Fuel", 2 July 2002.

(Jones, 2002a)
David Jones, "Sandia says its study not proof of nuke plant durability" Inside Energy, 30 September 2002, page 14.

(Jones, 2002b)
Gary Jones, Director, Natural Resources and Environment, US General Accounting Office, "Nuclear Waste: Uncertainties about the Yucca Mountain Repository Project", testimony before the Subcommittee on Energy and Air Quality, House Committee on Energy and Commerce, 21 March 2002.

(Keeny et al, 1977)
Spurgeon M. Keeny et al, Nuclear Power Issues and Choices (Cambridge, Massachusetts: Ballinger Publishing Company, 1977).

(Kot et al, 1982)
C. A. Kot et al, Evaluation of Aircraft Crash Hazards Analyses for Nuclear Power Plants, NUREG/CR-2859 (Washington, DC: NRC, June 1982).

(Krass, 1991)
Allan S. Krass, Consequences of the Chernobyl Accident (Cambridge, Massachusetts: Institute for Resource and Security Studies, December 1991).

(Kugler, 1995)
Richard L. Kugler, Toward a Dangerous World: US National Security Strategy for the Coming Turbulence (Santa Monica, California: RAND, 1995).

(Lange et al, 2002)
F. Lange et al, "Experiments to Quantify Potential Releases and Consequences from Sabotage Attack on Spent Fuel Casks", in Proceedings of the 13th International Symposium on the Packaging and Transportation of Radioactive Material (PATRAM) (Northbrook, Illinois: Institute of Nuclear Materials Management, 2002).

(Larson, 1985)
Jay R. Larson, Systems Analysis Handbook, NUREG/CR-4041 (Washington, DC: US Nuclear Regulatory Commission, November 1985).

(Lempert et al, 2000)
J. P. Lempert et al, "Status of HLW and Spent Fuel Management in the Federal Republic of Germany", paper presented at the WM'00 Conference, Tucson, Arizona, 27 February to 2 March 2000.

(Leventhal and Alexander, 1987)
Paul Leventhal and Yonah Alexander (editors), Preventing Nuclear Terrorism (Lexington, Massachusetts: Lexington Books, 1987).

(Macfarlane, 2001a)
Allison Macfarlane, "Interim Storage of Spent Fuel in the United States", Annual Review of Energy & Environment, Volume 26 (2001), pp 201-235.

(Macfarlane, 2001b)
Allison Macfarlane, "The problem of used nuclear fuel: lessons for interim solutions from a comparative cost analysis", Energy Policy, Volume 29 (2001), pp 1379-1389.

(Maguire, 2001)
Kevin Maguire, "Security checks tightened after high-level alert", The Guardian, 9 January 2001.

(Margulies and Martin, 1984)
T. S. Margulies and J. A. Martin, Dose Calculations for Severe LWR Accident Scenarios, NUREG-1062 (Washington, DC: NRC, May 1984).

(Markey, 2002)
Staff of US Representative Edward Markey, "Security Gap: A Hard Look At the Soft Spots in Our Civilian Nuclear Reactor Security", 25 March 2002.

(Martin, 2002)
John R. Martin (editor), Defeating Terrorism: Strategic Issue Analyses (Carlisle, Pennsylvania: Strategic Studies Institute, US Army War College, January 2002).

(Mathews, 2002)
Jessica T. Mathews, September 11, One Year Later: A World of Change (Washington, DC: Carnegie Endowment for International Peace, September 2002).

(McCough and Pennington, 2002)
Michael S. McCough and Charles W. Pennington, "Plane tough storage", Nuclear Engineering International, May 2002, pp 24-26.

(Meserve, 2002)
NRC Chairman Richard Meserve, "Statement Submitted by the United States Nuclear Regulatory Commission to the Committee on Environment and Public Works, United States Senate, Concerning Nuclear Power Plant Security", 5 June 2002.

(NAC, 2001)
NAC Worldwide Consulting, US Spent Fuel Update: Year 2000 in Review (Atlanta, Georgia: NAC Worldwide Consulting, 2001).

(National Intelligence Council, 2001)
National Intelligence Council, Foreign Missile Developments and the Ballistic Missile Threat Through 2015: Unclassified Summary of a National Intelligence Estimate (Washington, DC: Central Intelligence Agency, December 2001).

(National Research Council, 2002)
National Research Council, Making the Nation Safer: The Role of Science and Technology in Countering Terrorism (Washington, DC: National Academy Press, 2002).

(National Research Council, 1990)
National Research Council, Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V (Washington, DC: National Academy Press, 1990).

(NEI, 2002)
Nuclear Energy Institute, "Deterring Terrorism: Aircraft Crash Impact Analyses Demonstrate Nuclear Power Plant's Structural Strength", December 2002.

(Newhouse, 2002)
John Newhouse, "Assessing the Threats", in John Newhouse (editor), Assessing the Threats (Washington, DC: Center for Defense Information, July 2002).

(NRC, 2002)
US Nuclear Regulatory Commission, Fact Sheet: Nuclear Security Enhancements Since Sept. 11, 2001 (Washington, DC: US Nuclear Regulatory Commission, undated, apparently September 2002).

(NRC Staff, 2002)
"NRC Staff's Response to Contentions Submitted by San Luis Obispo Mothers for Peace et al", 19 August 2002, before the Atomic Safety and Licensing Board in the Matter of Pacific Gas & Electric Company, Diablo Canyon ISFSI, Docket No. 72-26-ISFSI, ASLBP No. 02-801-01-ISFSI.

(NRC, 1997)
US Nuclear Regulatory Commission, Standard Review Plan for Dry Cask Storage Systems, NUREG-1536 (Washington, DC: US Nuclear Regulatory Commission, January 1997).

(NRC, 1990)
US Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, NUREG-1150 (Washington, DC: US Nuclear Regulatory Commission, December 1990).

(Office of Homeland Security, 2002)
Office of Homeland Security, National Strategy for Homeland Security (Washington, DC: The White House, July 2002).

(Okrent, 1981)
David Okrent, Nuclear Reactor Safety: On the History of the Regulatory Process (Madison, Wisconsin: University of Wisconsin Press, 1981).

(Parry et al, 2000)
ASLBP No. 99-762-02-LA, "Affidavit of Gareth W. Parry, Stephen F. LaVie, Robert L. Palla and Christopher Gratton in Support of NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon which the Staff

Proposes to Rely at Oral Argument on Environmental Contention EC-6", 20 November 2000.

(Pennington, 2002)
Charles W. Pennington, "The Continuing Safety of NAC's Multipurpose Dry Storage Systems in the Face of Militant Acts of Destructiveness (MADness)", viewgraphs accompanying a presentation at the Institute of Nuclear Materials Management-sponsored Spent Fuel Management Seminar XIX, Washington, DC, 10 January 2002.

(Peters et al, 1999)
Ralf Peters et al, "CONSTOR - Spent Nuclear Fuel Storage and Transport Cask System", paper presented at WM'99 Conference, 28 February to 4 March 1999.

(PG&E, 2001a)
Pacific Gas and Electric Company, Diablo Canyon Independent Spent Fuel Storage Installation: Environmental Report (Avila Beach, California: PG&E, 21 December 2001).

(PG&E, 2001b)
Pacific Gas and Electric Company, Diablo Canyon Independent Spent Fuel Storage Installation: Safety Analysis Report (Avila Beach, California: PG&E, 21 December 2001).

(PLG, 1983)
Pickard, Lowe and Garrick Inc, Seabrook Station Probabilistic Safety Assessment, Main Report (Irvine, California: PLG, December 1983).

(POGO, 2002)
Project on Government Oversight, Nuclear Power Plant Security: Voices from Inside the Fences (Washington, DC: Project on Government Oversight, 12 September 2002).

(Powers, 2000)
Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, letter of 13 April 2000 to NRC Chairman Richard A. Meserve, subject: "Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants".

(Purdue, 2002)
Purdue University, "Purdue News: New simulation shows 9/11 plane crash with scientific detail", 10 September 2002.

(Raddatz and Waters, 1996)
M. G. Raddatz and M. D. Waters, Information Handbook on Independent Spent Fuel Storage Installations, NUREG-1571 (Washington, DC: US Nuclear Regulatory Commission, December 1996).

(Ramberg, 1984)
Bennett Ramberg, Nuclear Power Plants as Weapons for the Enemy: An Unrecognized Military Peril (Los Angeles: University of California Press, 1984).

(Raskin et al, 2002)
Paul Raskin et al, Great Transition: The Promise and Lure of the Times Ahead (Boston, Massachusetts: Stockholm Environment Institute, 2002).

(Rehm, 1984)
T. A. Rehm, memo to the NRC Commissioners, "Weekly Information Report -- Week Ending April 20, 1984".

(Resnikoff, 2001)
Marvin Resnikoff, Declaration of 30 January 2001 before the Atomic Safety and Licensing Board, US Nuclear Regulatory Commission, regarding the proposed Private Fuel Storage independent spent fuel storage installation (ASLBP No. 97-732-02-ISFSI).

(Reuters, 2002)
Reuters, "School says its missiles were licensed", The Boston Globe, 20 August 2002, page A11.

(Rufford et al, 2001)
Nicholas Rufford, David Leppard and Paul Eddy, "Nuclear Mystery: Crashed plane's target may have been reactor", The Sunday Times, London, 20 October 2001.

(Sailor et al, 1987)
V. L. Sailor et al, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982 (Washington, DC: NRC, July 1987).

(Scarry, 2002)
Elaine Scarry, "A nuclear double standard", Boston Sunday Globe, 3 November 2002, page D11.

(Shleien, 1983)
Bernard Shleien, Preparedness and Response in Radiation Accidents (Washington, DC: US Department of Health and Human Services, August 1983).

(SIPRI, 1981)
Stockholm International Peace Research Institute, Nuclear Radiation in Warfare (London: Taylor and Francis, 1981).

(Sloan, 1995)
Stephen Sloan, "Terrorism: How Vulnerable is the United States?", in Stephen Pelletiere (editor), Terrorism: National Security Policy and the Home Front (Carlisle, Pennsylvania: Strategic Studies Institute, US Army War College, May 1995).

(Sokolski and Riisager, 2002)
Henry D. Sokolski and Thomas Riisager, Beyond Nunn-Lugar: Curbing the Next Wave of Weapons Proliferation Threats from Russia (Carlisle, Pennsylvania: Strategic Studies Institute, US Army War College, April 2002).

(Sozen et al, 2002)
Mete A. Sozen et al, "September 11 Pentagon Attack Simulations Using LS-Dyna: Phase I, Completed September 11, 2002"
<http://www.cs.purdue.edu/homes/cmh/simulation/>

(Sprung et al, 2000)
J. L. Sprung et al, Reexamination of Spent Fuel Shipment Risk Estimates, NUREG/CR-6672 (Washington, DC: US Nuclear Regulatory Commission, March 2000).

(Stone and Webster, 1999)
Stone and Webster, Maine Yankee Independent Spent Fuel Storage Installation (ISFSI): Site Location of Development Permit #L-17973, Application for Amendment to Maine Department of Environmental Protection (Boston, Massachusetts: Stone and Webster, April 1999).

(Thompson and Beckerly, 1973)
T. J. Thompson and J. G. Beckerley (editors), The Technology of Nuclear Reactor Safety, Volume 2 (Cambridge, Massachusetts: MIT Press, 1973).

(Thompson, 2002a)
Gordon Thompson, Declaration of 7 September 2002 in support of a petition to the US Nuclear Regulatory Commission by Avila Valley Advisory Council et al, regarding nuclear-facility operations at the Diablo Canyon site.

(Thompson, 2002b)
Gordon Thompson, Civilian Nuclear Facilities as Weapons for an Enemy: A submission to the House of Commons Defence Committee (Cambridge, Massachusetts: Institute for Resource and Security Studies, 3 January 2002).

(Thompson, 2001)
Gordon Thompson, Declaration of 31 October 2001 in support of a motion by CCAM/CAM before the Atomic Safety and Licensing Board, US Nuclear Regulatory Commission, regarding the Millstone nuclear power station (ASLBP No. 00-771-01-LA).

(Thompson, 1999)
Gordon Thompson, Risks and Alternative Options Associated with Spent Fuel Storage at the Shearon Harris Nuclear Power Plant (Cambridge, Massachusetts: Institute for Resource and Security Studies, February 1999).

(Thompson, 1998a)
Gordon Thompson, "Science, democracy and safety: why public accountability matters", in F. Barker (editor), Management of Radioactive Wastes: Issues for local authorities (London: Thomas Telford, 1998).

(Thompson, 1998b)
Gordon Thompson, High Level Radioactive Liquid Waste at Sellafield: Risks, Alternative Options and Lessons for Policy (Cambridge, Massachusetts: Institute for Resource and Security Studies, June 1998).

(Thompson, 1996)
Gordon Thompson, War, Terrorism and Nuclear Power Plants (Canberra: Peace Research Centre, Australian National University, October 1996).

(Thompson and Sholly, 1991)
Gordon Thompson and Steven C. Sholly, No Restart for K Reactor (Cambridge, Massachusetts: Institute for Resource and Security Studies, October 1991).

(Throm, 1989)
E. D. Throm, Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools", NUREG-1353 (Washington, DC: US Nuclear Regulatory Commission, April 1989).

(Travers, 2002)
William D. Travers, memo to the NRC Commissioners, "Withholding Sensitive Homeland Security Information From the Public, COMSECY-02-0015", 4 April 2002.

(Travers, 2001)
William D. Travers, memo to the NRC Commissioners, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools (WITS 200000126), SECY-01-0100", 4 June 2001.

(Vermont Yankee, 2000)
Vermont Yankee Nuclear Power Corporation, Responses to CAN's 2nd Set of Interrogatories and Requests to Produce, 28 February 2000, Docket No. 6300, Response VY:CAN:2-21.

(Wald, 2001)
Matthew L. Wald, "US Failed to Learn From Earlier Hijackings", International Herald Tribune, 4 October 2001, page 6.

(Walker, 2002)
David M. Walker, Comptroller General of the United States, "Homeland Security: Responsibility and Accountability For Achieving National Goals", testimony before the Senate Committee on Governmental Affairs, 11 April 2002.

(Weyman, 1999)
Dean Weyman (Engineering Department, Vermont Yankee Nuclear Power Corporation), memo to Bob Jordan, 12 February 1999.

(White House, 2002)
The White House, The National Security Strategy of the United States of America (Washington, DC: The White House, September 2002).

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EXHIBIT RR

Procedure Contains NMM REFLIB Forms: YES ☐ NO ☒

Effective Date	Procedure Owner:	W. A. Eaton	Executive Sponsor:	W. A. Eaton
	Title:	VP Engineering	Title:	VP Engineering
12/1/2006	Site:	Echelon	Site:	Echelon

Exception Date*	Site	Site Procedure Champion	Title
	ANO	William G. Smith	Technical Specialist III
	GGNS	Bruce Lee	Technical Specialist IV
	IPEC	Richard Burroni	Manager, P&CE
	JAF	Joe Pechacek	Manager, P&CE
	PNPS	Gearld Bechen	Senior Engineer
	RBS	Reggie Jackson	Technical Specialist IV
	VY	Larry Lukens	Engineering Supervisor
	W3	Paul Stanton	Engineering Supervisor
	ECH	Charles Turk	Manager, Design Engineering
	WPO	Robert Penny	Manager, Engineering Programs

Site and NMM Procedures Canceled or Superseded By This Revision

- **ENN DC-315**
- **ENS-DC-315**

Process Applicability Exclusion)

All Sites: ☐ Specific Sites: ANO ☐ GGNS ☐ IPEC☐ JAF ☐ PNPS☐ RBS ☐ VY ☐ W3 ☐

Change Statement

Initial issue of fleet procedure which replaces ENN-DC-315 and ENS-DC-315.

*Requires justification for the exception

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1.0 PURPOSE

[P-33641 - P-33643], [P-33645], [P-33714 – P-33719], [P-33730], [P-35351], [JAFP-87-0737], [JPN-89-051], [IP3-87-055Z], [IPN-89-044], [BECo-89-107], [FVY-89-66], [FVY-87-94], [FVY-87-121], [P-1079], [P-35269], [P-24444], [P-15802], [P-15803], [P-16557], [P-20303], [P-22888]

[1] The purpose of this procedure is to provide requirements for establishing and maintaining an effective Flow Accelerated Corrosion (FAC) Program that will standardize Entergy Nuclear Fleet's approach towards mitigating FAC damage.

[2] This procedure uses a systematic approach for long term monitoring to enhance the reliability of the affected FAC components by reducing the probability of failures and reduces maintenance costs associated with unplanned or unnecessary repairs.

[3] This procedure provides criteria and methodology for selecting components for inspection, performing inspections, evaluating inspection data, disposition of results, sample expansion requirements, piping repair /replacement criteria, program responsibilities and documentation requirements.

[4] This program is applicable to carbon steel plant piping systems and includes feed water heater and moisture separator re-heater (MSR) shells susceptible to FAC. It includes inspections of single-phase and two-phase piping components for both safety and non-safety related systems.

[5] This procedure may be used as a guide for evaluating systems and components that are not included in the FAC program.

2.0 REFERENCES

[1] NRC Generic Letter 89-08, Erosion/Corrosion Induced Pipe Wall Thinning.

[2] NUREG-1344, “Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants.”

[3] NSAC 202L, latest revision, EPRI Document, "Recommendations for an Effective Flow Accelerated Corrosion Program”

[4] EPRI Technical Report, TR-106611, "Flow-Accelerated Corrosion in Power Plants"

[5] NRC Bulletin No. 87-01, "Pipe Wall Thinning."

[6] EN-LI-102, “Corrective Action Process.”

[7] Erosion/Corrosion in Nuclear Power Plant Steam Piping: Causes and inspection Program Guidelines. EPRI, April 1985. NP-3944.

2.0 cont.

[8] ENN-NDE-9.05,"Ultrasonic Thickness Examination"

[9] ANSI B31.1 "Power Piping", (For applicable code year see individual plant FSAR).

[10] ENN-DC-126, "Calculations".

[11] ENS-DC-126, "Engineering Calculation Process".

[12] ENS-DC-126-01, "Engineering Calculation Process".

[13] ENN-CS-S-008, "Pipe Wall Thinning Structural Evaluation".

[14] ENS-PS-S-001, "Pipe Wall Thinning and Crack-like Flaw Evaluation Standard".

[15] Site ASME XI Repair / Replacement Program as applicable.

[16] ENN-EP-S-005 "Flow Accelerated Corrosion Component Scanning and Gridding Standard".

[17] EPRI Paper, "Single-Phase Erosion/Corrosion of Carbon Steel Piping", February 1987.

[18] EPRI Paper - "Practical Consideration for the Repair of Piping Systems Damaged by Erosion/Corrosion", dated 10/5/87

[19] Acceptance Criteria for Structural Evaluation of Erosion/Corrosion Thinning in Carbon Steel Piping. EPRI, April 1988. NP-5911.

[20] NRC Generic Letter 90-05, "Guidance for Performing Temporary Non-Code Repairs of ASME Code Class 1, 2 &3 Piping".

[21] INPO SOER 87-3, "Piping Failures in High-Energy Systems Due to Erosion/Corrosion", March 1987.

[22] INPO Significant Operating Experience Report (SOER) 82-11, "Erosion of Steam Piping and Resulting Failure", February 1982.

[23] IN 93-21, Summary of NRC Staff Observations compiled during Engineering Audits on inspections of Licensee E/C Programs", dated March 25, 1993.

[24] EPRI CHUG Position Paper #3, "A Summary of Tasks and Resources Required to Implement an Effective Flow Accelerated Corrosion Program."

2.0 cont.

[25] EPRI CHUG Position Paper #4,"Recommendations for inspecting Feedwater Heater Shells for Flow-Accelerated Corrosion Damage", February 2000.

[26] CHECWORKS Steam /Feedwater Application, "Guidelines for Plant Modeling and Evaluation of Component Inspection Data", EPRI No. 1009599, Final Report, September 2004.

[27] Entergy Quality Assurance Manual

[28] ENN FAC Qualification Card ENN-TK-ESPG-042, "Implementing the Flow Accelerated Corrosion Program ".

[29] ENN/ENS-DC-115, "Engineering Response Development".

[30] EN-DC-115, Engineering Change Development"

[31] EOI-C-QC-ESPP-PFAC- "Qualification card for Flow Accelerated Corrosion Engineers".

[32] EN-DC-202, "NEI 03-08 Materials Initiative".

[33] JAF-SPEC-MISC-03290 Rev.0, "Specification for Evaluation and Acceptance of Local Areas of material, parts and components that are less than the specified thickness." By REEDY Engineering.

[34] IP3-SPEC-UNSPEC-02996 Rev.0, "Specification for Evaluation and Acceptance of Local Areas of material, parts and components that are less than the specified thickness." By REEDY Engineering.

[35] CHECWORKS Steam /Feedwater Application, Version 2.1, EPRI No. 1009600, Final Report, October 2004.

[36] CHECWORKS Steam /Feedwater Application, latest version.

[37] Institute of Nuclear Plant Operations, "Engineering Program Guide, Flow Accelerated Corrosion", EPG-06 (Pending).

3.0 DEFINITIONS

[1] Base Line Inspection – An initial wall thickness measurement of a component taken prior to being placed in service.

[2] Basis Document - Program documents that define the scope, attributes, commitments, evaluation reports and predictive models that forms the basis of the FAC program (i.e., System Susceptibility Evaluation reports). These documents contain the basis for the plant piping in the CHECWORKS model, the susceptible-not-modeled (SNM) piping and those that are non-susceptible.

[3] Code Minimum Thickness (t_{min}, $t_{codemin}$) – The minimum required global wall thickness based on hoop stress.

[4] Critical Thickness ($t_{crit.}$) -The minimum required wall thickness per code of construction required to meet all design-loading conditions.

[5] Deficient Component - A component identified by examination to be below t_{accpt} wall thickness or projected to be below t_{accpt} wall thickness by the next refueling outage.

[6] Degraded component – A component identified as being below the screening criteria that is acceptable for continued operation.

[7] EPRI CHUG – EPRI CHECWORKS USERS GROUP.

[8] Examination - Denotes the performance of all visual observation and nondestructive testing, such as radiography, ultrasonic, eddy current, liquid penetrant and magnetic particle methods.

[9] Examination Checklist/ Traveler – A data sheet developed for the components being inspected and may contain but is not limited to the following: t_{nom}, t_{meas}, Tmin, Screening criteria, components name, system number, previous data, inspection datasheet number, grid size, examination extent, work order and affiliated minimum wall calculation.

[10] Flow Accelerated Corrosion (FAC) - Degradation and consequent wall thinning of a component by a dissolution phenomenon, which is affected by variables such as temperature, steam quality, steam/fluid velocity, water chemistry, component material composition and component geometry. Previously known as Erosion/Corrosion.

3.0 cont.

[11] Grid - A pattern of points or lines on a piping component, where UT thickness measurements will be made. Grid may be permanently marked with circumferential and longitudinal grid lines.

[12] Grid Point – A Specific location on a piping component, where a UT thickness measurement will be made. Grid points are at the intersections of the circumferential and longitudinal grid lines.

[13] Grid Point Reading – UT reading taken at the intersection of the grid location.

[14] Grid Scan– 100% scans of the area between the grid lines. The lowest measurement in each area to be recorded as the measured thickness.

[15] Full Scan – scans of 100% of an area, circumference, nozzle, heater segments etc, measuring minimum, maximum and averages thicknesses and approximate location of minimum measured thickness.

[16] Grid Size - The distance between grid points in the circumferential or longitudinal direction. Also called grid space or grid spacing.

[17] Initial Thickness (t_{init}): The thickness determined by ultrasonic examination prior to the component being placed into service (baseline) or the first ultrasonic examination during its service life. If an examination has not previously been performed on the component, the initial thickness shall be determined by reviewing the initial ultrasonic data for that component. The area of maximum wall thickness within the same region as the worn area (based on the method selected for evaluating wear) shall be identified and compared to t_{nom}. If the thickness is greater than t_{nom}, the maximum wall thickness within that region shall be used as t_{init}. If that thickness is less than t_{nom}, t_{nom} shall be used as t_{init}.

[18] Inspection Location - A specific component (i.e., elbow, tee, reducer, straight pipe section).

[19] Inspection Outage - the outage during which the component was inspected.

[20] Large-bore Piping - Piping generally greater than 2" nominal pipe size with butt-weld fittings.

[21] Line Scans– piping segments broken into one-foot lengths (Small-Bore pipe).

3.0 cont.

[22] Minimum acceptable wall thickness (t_{accpt}) – Maximum value of axial stress, hoop stress, and or critical thickness and the piping replacement values of 0.3 t_{nom} for Class1 piping or 0.2 t_{nom} for Class 2, Class 3 and non-safety related piping.

[23] Minimum Measured Thickness - (t_{meas} or t_{mm}) as identified by ultrasonic thickness examination, the present thickness at the thinnest point on a component.

[24] Local minimum required thickness – (t_{aloc}) Minimum acceptable local wall thickness as calculated by ENN-CS-S-008 or ENS-PS-S-001.

[25] Minimum required thickness – (t^{a}_{min}) Minimum required pipe wall thickness based on axial stress (See ENN-CS-S-008) .

[26] Next Scheduled Inspection (NSI) -The outage at which an inspection will be performed on a given component.

[27] Nominal Thickness (t_{nom}) - Wall thickness equal to ANSI standard thickness.

[28] PASS 1 Analysis - Runs modeled in CHECWORKS that either have no inspection data, an insufficient number of inspections to provide a proper calibration, or where there is no expectation of ever developing a proper calibration.

[29] PASS 2 Analysis - The process of utilizing UT inspection data thickness measurements in CHECWORKS to predict wear and wear rates for components.

[30] Piping Segment - A run of piping that consists of inspection locations which have common operating parameters (i.e., temperature, pressure, flow rate, Oxygen content and pH level).

[31] Predicted /Projected Thickness (t_p, t_{pred}) -The calculated thickness of a component based upon a rate of wear to some point in time (e.g., next refueling, next scheduled examination).

[32] Quadrant Scan– Piping segments divided in quadrants A, B, C, D that are 90 degrees apart and broken into one-foot lengths, or as specified by the FAC engineer.

3.0 cont.

[33] Qualified FAC Engineer- Individual who has completed the FAC Qualification Card, who participates in the Engineering Support Personnel (ESP) training program and demonstrates knowledge required for the use of the CHECWORKS computer program.

[34] Reference Point - The point on a piping component where the longitudinal and circumferential grid lines originate.

[35] Remaining Service Life (RSL) - The amount of time remaining based upon an established rate of wear at which the component is anticipated to thin to t_{accpt}.

[36] Safety Factor – A Margin of Safety used to account for inaccuracies in wear rate evaluation.

[37] Sample Expansion - The addition of inspection locations based on significant or unexpected wall thinning during planned inspection(s).

[38] Significant wall thinning - Wall thinning to a thickness which is the largest of:

(a) a thickness less than 60% of pipe nominal wall thickness

(b) Wall thinning to a thickness that is half the remaining margin of the piping/ component which is above t_{accpt}. $[½ (0.875\ t_{nom} + t_{accpt})]$

(c) $(t_{accpt} + 0.020)$ inch.

[39] Small-bore Piping - Piping that is generally 2" or less nominal diameter and that typically uses socket welded fittings.

[40] Subsequent Inspection - Inspection of components that have had a baseline inspection and/or an initial operational inspection.

[41] Susceptible Line - Piping determined to be susceptible to FAC using the EPRI susceptibility criteria in NSAC 202L, industry experience and as documented in the System Susceptible Evaluation.

[42] Susceptible Non-Modeled (SNM) Piping - A subset of the FAC susceptible lines that cannot be modeled using the EPRI CHECWORKS software.

[43] Time - Time in service shall be actual hours on line or of operation and/ or hours critical. Calendar hours may be used for conservatism.

3.0 cont.

[44] Train – Loops within subsystems that have similar geometries, flow rates and temperatures and which have similar FAC risk.

[45] UT Datasheets - Paperwork that documents the results of the ultrasonic thickness inspections.

[46] Wear (W) - The amount of material removed or lost from a components wall thickness since baseline or subsequent to being placed in service.

[47] Wear Rate (WR) - Wall loss per unit time.

4.0 RESPONSIBILITIES

4.1 MANAGER, ENGINEERING PROGRAMS

[1] Providing a single point of accountability and is responsible for the overall health and direction of the FAC programs.

[2] Ensuring that the FAC programs are effectively developed and implemented.

[3] Providing oversight for implementing the FAC programs.

[4] Co-ordinate ENN or ENS FAC working group meetings.

[5] Co-ordinate ENN or ENS FAC Self-Assessments.

4.2 SUPERVISOR, CODE PROGRAMS

[1] Designate responsible engineer/Personnel from the Code Programs Engineering Group for the implementation and maintenance of the Flow Accelerated Corrosion Program.

[2] Ensure that the Flow Accelerated Corrosion Program activities are conducted in accordance with this procedure.

[3] Shall ensure that repair procedures are in place to support any planned repairs or replacements.

[4] Ensure audits and surveillance of selected Flow Accelerated Corrosion (FAC) activities is performed to verify compliance with applicable codes, procedures and drawings.

4.2 cont.

[5] Provides personnel to perform NDE during normal plant operation and unscheduled outages.

[6] Shall provide qualified Non-Destructive Examination personnel to perform flow accelerated corrosion inspections during scheduled refueling and maintenance outages.

[7] Provides personnel to perform reviews of all final FAC UT data sheets.

[8] Provides personnel to review vendor procedures, personnel certifications and equipment certifications.

[9] Assuring adequate technical personnel are available to provide required support services prior to the outage.

[10] Allocation of resources to execute the requirements of the program.

[11] Provide funding and resources to address control and configuration requirements for FAC drawings.

[12] Having bench strength and back up personnel for the FAC program.

4.3 NDE LEVEL III OR DESIGNEE

[1] Reviews and approves FAC personnel and equipment certifications, and NDE procedures including revisions.

[2] NDE Level II or Level III reviews and signs all final NDE/UT data sheets to ensure appropriate NDE examinations have been completed in accordance with the FAC program. The NDE level III review of Risk Informed examination shall be performed in accordance with the site ISI program requirements.

[3] Resolution of anomalies found in inspection data.

[4] Identify discrepancies or deficiencies and initiates condition report in accordance with FAC program or site protocols as appropriate.

[5] Performs oversight of selected FAC examinations to verify vendor procedure compliance.

[6] Performs functions in accordance with applicable procedures including the Entergy Quality Assurance Program.

4.4 FLOW ACCELERATED CORROSION ENGINEER

[1] Shall determine scope of inspections. The FAC Engineer shall develop a list of components/piping segments to be inspected prior to each outage using the criteria of NSAC-202L and CHECWORKS Pass1 and Pass 2 output as a guide. Previous outage inspection results shall be reviewed prior to development of the inspection list. This list shall be based on the susceptibility to flow accelerated corrosion and the severities of wear identified from previous inspection results.

[2] Review and/or perform an engineering evaluation for all Flow Accelerated Corrosion inspections where pipe wall thinning has been identified and concur on any recommended action. Calculations shall be done in accordance with applicable procedures.

[3] Shall ensure that appropriate inspections are performed in accordance with the scope of the Flow Accelerated Corrosion Program.

[4] Shall review and may sign all inspection data and make recommendations for repair/replacement of piping materials in accordance with applicable site protocols.

[5] Shall provide NDE data for review and signature to the ANII, if requested by the ANII.

[6] Shall provide Risk Informed Inspection data sheet (s) to the ANII for review and signature, if applicable.

[7] Develops or reviews program basis documents.

[8] Shall revise and/or expand the scope of the Flow Accelerated Corrosion inspection program to incorporate industry and in-house operating experiences and track/trend inspection results.

[9] Shall maintain records of all inspection results and inspection database.

[10] Develop a FAC examination checklist/traveler that contains t_{nom}, screening criteria, t_{accpt}, line number, etc. for the components being inspected.

[11] Shall initiate request for engineering services in accordance with the MAXIMO/Indus Asset Suite or site specific work control system for piping replacement or engineering evaluations as required. This request should include recommended materials for replacement and configuration changes, if applicable, to reduce the effects of flow accelerated corrosion.

[12] Shall periodically review completed plant modifications to assess their effect on the scope of the flow accelerated corrosion program.

4.4 cont.

[13] Shall assist in vendor oversight as required.

[14] Maintaining control of the predictive models (e.g. CHECWORKS), which includes any development, updates or revisions to the models.

[15] Developing, revising, and issuing FAC program documents.

[16] Initiating and/or responding to Condition Reports and Engineering Requests for evaluating degraded and deficient components or other discrepancies or deficiencies within the scope of the FAC program.

[17] Developing post outage inspection summary reports.

[18] Review and disposition Operating Event (OE) notices for applicability to the FAC program.

[19] Analyzing inspection data to determine component acceptability for continued service and to determine the need for sample expansion.

[20] Prioritizing and ranking inspection in terms of susceptibility and consequence of failure.

[21] Develop and maintain the System Susceptibility Evaluation report.

4.5 DESIGN ENGINEERING/RESPONSIBLE ENGINEER

[1] Provide minimum acceptable wall thickness (t_{accpt}) to the FAC Engineer. Responsibility may be delegated to another department or qualified personnel.

[2] Perform local wall thinning evaluations for components having UT measurements that are below or are projected to go below the minimum acceptable wall thickness (t_{accpt}) or administrative wall thickness requirement. Responsibility may be delegated to another department or qualified personnel.

[3] Prepare and issue engineering response packages for component requiring replacement. Responsibility may be delegated to another department or qualified personnel.

[4] Perform remaining service life evaluation for components in the FAC program as required. Responsibility may be delegated to another department or qualified personnel.

4.6 MAINTENANCE SUPERVISOR/DESIGNEE

[1] The maintenance supervisor or designee will ensure that adequate craft personnel are available to support the FAC program. The supervisor shall ensure that scaffolding is erected, when needed, and insulation removed from components/piping segments that will be inspected and that the piping is prepared for inspection. Scaffolding erection in safety related areas should be in accordance with site procedures.

[2] The maintenance supervisor or designee shall inform the FAC engineer when it is necessary to remove a pipe support for inspection. An engineering evaluation is required if a pipe support requires removal.

[3] The maintenance supervisor must ensure that surfaces to be inspected are free from all foreign materials that would interfere with the inspections, i.e., dirt, rust, paint, etc. If cleaning is required, this may be accomplished by power sanding, flapper wheel only) hand wire brushing, or hand sanding in accordance with site procedures/protocols.

[4] The maintenance supervisor shall ensure restoration of lines, i.e. insulation replaced, scaffolding removed, upon completion of the FAC inspection.

4.7 FAC/ISI PROJECT COORDINATOR

[1] A FAC/ISI project coordinator may be chosen to Implement the activities of the inspection plan, the duties, if applicable, may include but is not limited to the following activities:

(a) Performing component walk downs

(b) Generating NDE inspection packages

(c) Defining NDE staffing as required

(d) Scheduling of inspections

(e) Acquiring data as required

(f) Providing field coordination to ensure timely inspection are accomplished

(g) Tracking progress of the FAC inspection project

(h) Transmitting inspection results to the FAC Engineer

5.0 DETAILS

5.1 PRECAUTIONS AND LIMITATIONS

None.

5.2 ANALYSIS/PRE-EXAMINATION

[1] The criteria contained in NSAC-202L, latest revision, shall be used to perform the System Susceptibility Evaluation (SSE).

[2] The System Susceptibility Evaluation report shall be developed and peer checked in accordance with ENN or ENS procedures.

[3] Non-typical operation of systems should be taken into consideration and if necessary factored into the FAC program.

[4] The susceptible small-bore piping inspection priority ranking should consider personnel safety, consequence of failure and plant unavailability.

[5] Industry and plant experiences relating to FAC will be factored into the program.

[6] The CHECWORKS model should be used for guidance in determining inspection priority based on relative ranking for specific locations to be examined for FAC damage.

5.3 PREPARATION OF OUTAGE INSPECTION PLAN

[1] The FAC Program Engineer shall prepare an Outage Inspection Plan prior to the outage to meet site milestones.

[2] The Outage Inspection Plan should consider the cost of repair/replacement versus inspection.

[3] The Outage Inspection Plan should consider inspection priority based on relative ranking for specific locations to be examined for FAC damage.

[4] Each identified location shall be documented in the inspection plan, along with the component number and reason for selection.

[5] The inspection plan shall be reviewed by qualified FAC personnel.

5.3 cont.

[6] Component Selection

(a) The FAC engineer shall prepare a FAC Outage Inspection scope as directed by plant milestones or as directed by Station management.

(b) Inspection selections shall be made in accordance with the requirements of this procedure and shall be identified based on CHECWORKS results, industry/station/utility experience, required re-inspections, the non- modeled program piping and engineering judgment.

(c) If a selected inspection location is determined to be excessively difficult, impractical or costly to examine due to inaccessibility, temperature, ALARA concerns, scaffolding requirements, or other factors, then an equivalent alternate inspection location may be selected.

(d) Components selected shall be formally documented.

(e) The criteria for component selection should consider the following:

(1) Components selected from measured or apparent wear found in previous inspection results.

(2) Components ranked high for susceptibility from current CHECWORKS evaluation.

(3) Components identified by industry events/experience via the Nuclear Network or through the EPRI CHUG.

(4) Components selected to calibrate the CHECWORKS models.

(5) Components subjected to off normal flow conditions. Primarily isolated lines to the condenser in which leakage is indicated from the turbine performance monitoring system.

(6) Engineering judgment / Other

(7) Piping identified from Work Orders (malfunctioning equipment, downstream of leaking valves, etc.).

(8) Susceptible piping locations (groups of components) contained in the Small Bore Piping database, which have not received an initial inspection.

5.3[6](e) cont.

(9) Piping identified from Condition Reports/ Corrective action, Work Orders (malfunctioning equip, downstream of leaking valves, etc.).

(10) Vessel Shells – Feed-water heaters, moisture separator re-heaters, drain tanks etc.

[7] Inspection schedule

(a) Inspection sequence and schedule should be developed based on priority established by the FAC engineer considering repair/scope expansion potential. Consideration will also be incorporated based on other outage work priorities, job conflict and system window duration.

(b) The FAC outage schedule should contain sufficient time for analysis and evaluations of the components being inspected.

[8] Drawing Preparation

(a) For each component scheduled for inspection, an isometric or other acceptable location drawing should be prepared prior to the outage that identifies the component to be examined. When applicable ensure the component number is shown on the drawing.

[9] Obtain Minimum Acceptable Wall Thickness (t_{accpt})

(a) Obtain t_{accpt} values for each component to be inspected.

(b) The minimum acceptable wall thickness, t_{accpt}, values should be obtained from ENN-CS-S-008 or ENS-PS-S-001 as applicable or from an approved site method <u>(e.g. FAC Manager).</u>

(c) Values for t_{accpt} should be obtained from design engineering or it may be delegated to another department or qualified personnel. These values may be ascertained prior to or during an outage.

5.3 cont.

[10] Component Identification

(a) Inspected components should have a unique identifier to allow for the tracking of inspection data.

(b) Component identifiers may allow for the identification of the Unit, system, sub-system, line number and corresponding location of that component within a sub-system.

(c) Components in the CHECWORKS non-modeled piping may be identified by using line numbers.

[11] Pre-inspection Activities

(a) Review inspection schedule, inspection requirements and sequence with appropriate plant personnel to ensure requirements for the completion of the FAC inspection are understood.

(b) The FAC engineer should participate in the preparation of FAC inspection work packages as required.

5.4 GRIDDING

[1] Gridding of components shall be performed in accordance with recommendation of NSAC 202L, ENN-EP-S-005 (for ENN plants only), and applicable site approved procedures or as specified by the FAC engineer.

[2] Gridding information shall be documented on the appropriate NDE UT data sheet either by a sketch or digital photo.

5.5 NDE TEST METHODS AND DOCUMENTATION

[1] Components can be inspected for FAC wear using ultrasonic testing (UT), radiography testing (RT), visual observation or other approved methods. The inspection technique used shall be at the discretion of the FAC engineer.

5.5 cont.

[2] UT thickness measurement is the primary method of determining pipe wall thickness.

(a) Inspections will be performed by using one of the following techniques:

(1) Grid Point Reading

(2) Grid Scan

(3) Quadrant Scan

(4) Line Scan

(5) Full Scan

(b) Ultrasonic Thickness measurement shall be performed in accordance with approved NDE, site or vendor procedures.

(c) A data sheet for components inspected shall be prepared. The information included in the sheet should contain but is not limited to the following:

(1) Plant's name/unit

(2) Components name

(3) Component sketch

(4) NDE technician signature/ date

(5) Grid size

(6) Axial and radial grid boundaries

(7) Calibration information

(8) Level II or Level III signature/date

(9) Work order information

(10) Nominal & Measured thickness

(11) 87.5% nominal thickness screening criteria

(12) Scanning method

5.5 cont.

[3] Radiograph Testing

(a) RT (digital or conventional) is the preferred method for inspecting socket welded fittings. The method used is at the discretion of the FAC engineer.

(b) RT can be performed during plant operations without removing insulation

[4] Visual Observation

(a) Visual observation/techniques may be used for examination of large components such as tanks, cross-around piping, cross-under piping, pump casings, shell walls, valves etc. (visual techniques is only applicable to two phase flow).

(b) Follow-up UT examinations, at the discretion of the FAC engineer, may be required of areas where significant damage is observed or suspected.

5.6 EVALUATION OF UT INSPECTION DATA

NOTE
Historically, typical manufacturing practice has been to supply fittings (especially tees, elbows and reducers) with wall thickness significantly larger than the piping nominal thickness.

[1] The data review should consider screening for further evaluation. Factors that should be considered when reviewing the inspection data include unknown initial thickness (especially fittings), counter-bore, obstructions, and manufacturing wall thickness variations.

[2] For each component that is examined and is below the screening criteria of 87.5% of nominal wall, the wear, wear rate, remaining service life shall be calculated.

[3] The FAC Program Engineer or designee shall review the UT data to ensure that the data is complete and corresponds to the requirements specified on the inspection data sheet (i.e., grid size, spacing, flow direction, starting and ending locations, obstructions, missing data, suspect readings and orientation).

[4] If low readings are encountered from repeat inspections that are due to counter-bore, then those areas shall be noted and additional inspections are not required.

5.6 cont.

[5] Grid Refinement

(a) A grid reduction / refinement may be used if the minimum measured thickness is less than the minimum required wall thickness, severe wall thinning is detected, engineering judgment, or the projected thickness is less than the minimum required wall thickness or as directed by the FAC engineer.

(b) The results of the grid refinement or scan shall be documented on an inspection data sheet.

[6] Grid Extension

(a) If measurement indicates wall loss at any edge of the grid, then the grid should be extended until the entire wear pattern is mapped.

[7] Determination of Initial Wall Thickness

(a) Initial Thickness (t_{init}): The thickness determined by ultrasonic examination prior to the component being placed into service (baseline) or the first ultrasonic examination during its service life. If an examination has not previously been performed on the component, the initial thickness shall be determined by reviewing the initial ultrasonic data for that component. The area of maximum wall thickness within the same region as the worn area (based on the method selected for evaluating wear) shall be identified and compared to t_{nom}. If the thickness is greater than t_{nom}, the maximum wall thickness within that region shall be used as t_{init}. If that thickness is less than t_{nom}, t_{nom} shall be used as t_{init}.

[8] Determination of Wear

(a) Wear of piping components may be evaluated using the band, area, and blanket or point-to-point method as defined in NSAC-202 L, latest revision or any other approved method.

(b) Evaluation of inspection data that is determined to require wear evaluation shall be documented and reviewed.

5.6 cont.

[9] Wear rate Determination

(a) Wear rate is determined by wear/ unit time (Units to be consistent with thickness evaluation).

(b) A reasonable safety factor should be applied to the wear rates to account for inaccuracies in the FAC wear rate calculations.

(c) Wear rate evaluation should be evaluated on a component evaluation sheet.

[10] Predicted Thickness (t_p , t_{pred})

(a) The projected or predicted thickness to the next schedule refueling outage.

$$t_{pred} = t_{meas} - \text{Safety factor x Wear Rate x Time}$$

A safety factor of 1.1 should be applied to all Entergy nuclear plants. If a value less than 1.1 is used the reason shall be documented.

[11] Determination of Remaining Service Life (RSL)

(a) Remaining service life (RSL) shall be evaluated as follows, units to be consistent with thickness evaluation:

$$RSL = (t_{meas} - t_{accpt}) / (\text{Safety Factor x Wear Rate})$$

5.7 EVALUATION OF RT INSPECTION DATA

[1] Qualified NDE personnel shall interpret the film and report the examination result to the FAC engineer.

[2] Appropriate conservatism should be used to determine if a component requires replacement or re-inspection as a consequence of qualitative nature of RT.

[3] RT inspection shall be recorded on a data sheet.

5.8 EVALUATION OF VISUAL INSPECTION DATA

[1] Where accessible, visual inspections may be performed on two-phase flow lines.

[2] Follow-up UT inspection is required for locations where significant damage is observed or suspected.

[3] Due to the qualitative nature of visual inspections, appropriate conservatism should be used when determining whether a component is acceptable to return to service and when establishing a re-inspection frequency.

5.9 DISPOSITION OF INSPECTION RESULTS

[1] The following are used to disposition component inspection results. Reference attachment 9.3 for logic diagram

> NOTE
>
> Certain components may have very little margin remaining as a consequence of high stresses in the line even though $t_{pred} \geq 0.875\ t_{nom}$ and therefore may require evaluation, for example Feedwater, Condensate, RHR, etc.

[2] If t_{pred} is $\geq 0.875\ t_{nom}$, the component is acceptable as is and may be returned to service.

[3] If t_{pred} is $< 0.875\ t_{nom}$, evaluate for sample expansion (Reference section 5.12).

[4] If t_{pred} is $\leq 0.3\ t_{nom}$, for ISI Class 1 piping repair or replacement is required in accordance with the requirements of ASME Section XI Repair and Replacement Program.

[5] If t_{pred} is $\leq 0.2\ t_{nom}$, for ISI Class 2, Class 3 and non-safety related, repair, replace or evaluate as warranted in accordance with applicable site programs or as directed by the FAC engineer.

[6] If t_{pred} is $\geq t_{accpt}$, the component is acceptable for continued operations, however monitoring is required in accordance with program requirements.

5.9 cont.

[7] If t_{pred} is < t_{accpt}, a structural evaluation is required in accordance with site approved procedures or engineering standards. Also a sample expansion evaluation is required. Repair or replacement in accordance with the requirements of ASME Section XI Repair and Replacement Program or other site approved process may also be required.

[8] If t_{meas} is < t_{accpt}, **generate a condition report**. A structural evaluation is also required in accordance with applicable site procedures or engineering standards.

5.10 RE-INSPECTION REQUIREMENT

[1] If the remaining service life (RSL) of a component is greater than or equal to the number of hours in the next operating cycle, then the component may be returned to service.

[2] If the component's remaining service life (RSL) is greater than the number of hours in the next operating cycle but is less than the number of hours in the next two operating cycles, then the component should be considered for re-inspection, repair or replacement during the next scheduled outage.

[3] If the component is acceptable for continued service, then it shall be re-examined before or during the outage immediately prior to the cycle during which it is projected to wear to the minimum allowable wall thickness.

5.11 COMPONENTS FAILING TO MEET INITIAL SCREENING CRITERIA

[1] If the results of the remaining life evaluation are shorter than the amount of time until the next scheduled inspection, there are several options for disposition of the component, as follows:

(a) Shorten the inspection interval (for components that can be inspected online)

(b) Refine the t_{accpt} value through a detailed stress analysis, which should be provided by Design Engineering or designee.

(c) Repair or replace the component

(d) ISI Class1 components that are less than or equal to 0.3 t_{nom} must be repaired or replaced unless further structural evaluation permits continued service.

[2] Wall thinning resulting in less than t_{accpt} shall be reported immediately to the FAC engineer by verbal or written communications.

5.11 cont.

[3] A condition report shall be generated when significant wall thinning or unexpected wear is detected in a system or component.

[4] A condition report shall be generated for wall thinning below t_{accpt} or other site established limit and a subsequent structural evaluation performed to disposition the line for continued service.

[5] If a previous condition report was generated for a component with wall thinning then no new condition report is required provided that the associated structural evaluation is current and applicable.

5.12 SAMPLE EXPANSION

[1] If a component is discovered that has a current or projected wall thickness less than the minimum acceptable wall thickness (t_{accpt}), then additional inspections of identical or similar piping components in a parallel or alternate train shall be performed to bound the extent of thinning except as provided below. Reference section 5.12.2.

[2] When inspections of components detects significant wall thinning and it is determined that sample expansion is required, the sample size for that line should be increased to include the following:

(a) Components within two diameters downstream of the component displaying significant wear or within two diameters upstream if the component is an expander or expanding elbow.

(b) A minimum of the next two most susceptible components from the relative wear ranking in the same train as the piping component displaying significant wall thinning.

(c) Corresponding components in each other train of a multi-train line with a configuration similar to that of the piping component displaying significant wall thinning.

[3] If the expanded inspection scope detects additional degradation, the sample expansion should continue until no additional components with significant wear are detected.

[4] Sample expansion is not required if the thinning was expected or if the thinning is unique to that component (e.g., degradation downstream of a leaking valve).

5.12 cont.

[5] Inspections of components from the current or past outages may satisfy the sample expansion criteria, therefore, some of the sample expansion requirements can be met without performing additional inspections.

[6] Sample expansion is not required for components that are being re-inspected if normal or expected wear is detected or wear unique to that component. All other wear patterns encountered shall be evaluated by the FAC Engineer to determine if sample expansion is required.

5.13 REPAIR / REPLACEMENT OF DEGRADED COMPONENTS
[NRC Generic Letter 90-05]

[1] The FAC engineer shall generate applicable documents to facilitate repair or replacement of degraded or deficient components.

[2] Components experiencing severe or unacceptable wear should be replaced with corrosion resistant material. However like in kind replacement may be appropriate if procurement of a resistant material would delay plant restart.

[3] Replacing components or fitting-by-fitting that have experienced significant wear is a satisfactory approach to reducing wear if the wear is very localized (i.e., wear is concentrated downstream of a flow control valve or orifice).

[4] Repairs and replacements to piping and components within the scope of Class 1, 2, 3 shall be performed in accordance with the requirements of ASME Section XI Repair and Replacement Program.

[5] All temporary non-code repairs to ISI Class 1, 2, 3 shall comply with NRC Generic Letter 90-05.

5.14 COMPONENT EVALUATION PACKAGES

[1] The FAC Engineer or designee shall assemble a component evaluation package for each examined component which may contain some of, but is not limited to the following:

(a) UT DATA Sheet

(b) Isometric drawing(s), sketches, flow diagram and digital photo.

(c) Reference to Structural /Minimum wall evaluation

(d) Component evaluation data sheet.

5.15 POST- INSPECTION ACTIVITIES

[1] The FAC Program Engineer shall prepare an Outage Summary report to document the outage FAC activities and submit to Records for retention in accordance with applicable procedures.

[2] Update CHECWORKS models with inspection data.

[3] Update small bore susceptible report as applicable

[4] Update all applicable FAC reports.

[5] Update FAC System Susceptible Report as required.

5.16 LONG TERM STRATEGY

[1] Entergy's fleet long-term strategy shall focus on reducing the plants FAC susceptibility. Optimization of the inspection planning process is an important factor. However, the reduction of FAC wear rates is necessary if both the number of inspections and the probability of failure are to be reduced. Subsequently the fleet's long term strategy will include the following elements:

(a) The use of improved materials for replaced components or proactive replacement of piping with corrosion resistant material.

(b) Utilization of improved water chemistry

(c) Incorporation of local design changes.

(d) Optimization of the inspection planning process,

(e) Industry participation in meetings for technology and information transfer (e.g. EPRI CHUG).

(f) Maintaining up-to-date predictive software and incorporating the latest inspection data in the models.

5.17 METHODS OF DETERMINING PLANT PERFORMANCE

[1] Program performance indicators, self- assessments and bench marking are utilized as methods for monitoring program and plant performance.

6.0 INTERFACES

[1] ENN-CS-S-008, “Pipe Wall Thinning Structural Evaluation”.

[2] ENN-EP-S-005 "Flow Accelerated Corrosion Component Scanning and Gridding Standard".

[3] ENS-PS-S-001, “Pipe Wall Thinning and Crack-like Flaw Evaluation Standard”.

[4] EN-DC-202, “NEI 03-08 Materials Initiative”.

7.0 RECORDS

[1] Record retention shall be in accordance with site procedures.

8.0 OBLIGATION AND COMMITMENTS IMPLEMENTED BY THIS PROCEDURE

8.1 OBLIGATIONS AND COMMITMENTS IMPLEMENTED OVERALL

None

8.2 SECTION/STEP SPECIFIC OBLIGATIONS AND COMMITMENTS

Step	Document	Document Section/Step	Commitment Number
[1]	QAPM	A.6a, A.6b, A.6c, A.6e	P33641–P33643, P-33645
[2]	QAPM	B.12a, B.12b, B.12c, B.12d, B.12e, B12f	P-33714 – P-33719
[3]	QAPM	B.15a, B.15c	P-33730, P-35351
[4]	NRC Generic Letter 90-05		None

8.3 SITE SPECIFIC COMMITMENTS

Step	Site	Document	Commitment Number or Reference
[1]	JAF	Response to NRC IE Bulletin 87-01	JAFP 87-0737
[2]	JAF	Response to NRC Generic Letter 89-08	JPN-89-051
[3]	IPEC Unit 3	Response to NRC IE Bulletin 87-01	IP3-87-055Z
[4]	IPEC Unit 3	Response to NRC Generic Letter 89-08	IPN-89-044
[5]	IPEC Unit 2	Response to NRC IE Bulletin 87-01	Mr. Murray Selman (Con Edison) to Mr. William Russell (NRC), Letter dated September 11, 1987.

8.3 cont.

[6]	Pilgrim	Response to NRC Generic Letter 89-08	BECo 89-107
[7]	VY	Response to NRC Generic Letter 89-08	Vermont Yankee letter to USNRC, FVY-89-66
[8]	VY	Response to NRC IE Bulletin 87-01	Vermont Yankee letter to USNRC, FVY-87-94
[9]	VY	Supplemental Response to NRC IE Bulletin 87-01	Vermont Yankee letter to USNRC, FVY-87-121
[10]	ANO	OCAN108914	P-1079
[11]	GGNS	GGNS Appendix K, Power Uprate	P-35269
[12]	GGNS	Response to NRC Generic Letter 89-08	P-24444
[13]	RBS	Response to NRC IE Bulletin 93-02	P-15802
[14]	RBS	Response to NRC IE Bulletin 93-02, Supp. 1	P-15803
[15]	WF3	Response to INPO SOER 87-03	P-16557
[16]	WF3	Response to IEN 89-001	P-20303
[17]	WF3	Response to IEN 93-021	P-22888

(a)

9.0 ATTACHMENTS

9.1 Guidance on Parameters affecting FAC.

9.2 Flow Accelerated Corrosion Program Attributes.

9.3 Wall Thinning Evaluation Process Map.

GUIDANCE ON PARAMETERS AFFECTING FAC

Listed below are factors to be considered when reviewing work requests, component replacements and modification packages for possible impact on the content of the FAC Program governed by DC-315. All Design Change Packages (DCP's) are required to be evaluated for impact to the FAC Program. This list is not intended to be all-inclusive or to limit the number of items an individual would consider when performing this impact assessment. It is intended as a reasonable list of items to consider for potential program content updates.

1. Water Chemistry. Many water chemistry parameters have been shown to contribute to FAC.

   a. pH Control Amine – pH is the primary chemistry parameter affecting FAC rates in PWRs. However, the amine used to control pH also plays an important role. Amines such as ammonia tend to separate more into the steam phase in two-phase flow conditions, and therefore provide less protection in the drains. Amines such as morpholine and especially ethanolamine have better partitioning characteristics for FAC.

   b. In a BWR, pH has much less of a role since the pH is stable and there are no amine's added to control the pH. FAC rates decrease as pH level increases. FAC rates seem to drop considerably at pH values of greater than 9.3 - 9.5.

   c. Oxygen Content - FAC rates decrease as oxygen concentration increases. Values that typically result in minimum FAC rates are approximately 15 to 20 ppb.

   d. Hydrogen Water Chemistry – BWR Plants that do not have hydrogen addition normally have a main steam oxygen content near 18 ppm. Plants with hydrogen water chemistry typically have an oxygen content from 3 to 12 ppm. This has a potential to impact the corrosion rates in the LP steam systems; mainly the first and second stage reheater drains based on industry experience.

   e. Hydrazine Injection - Hydrazine is added to the feed train of PWRs as an oxygen scavenger and to maintain a reducing environment in the steam generators. From zero to approximately 150 ppb,an increase in hydrazine concentrations seems to increase rates of FAC. Higher concentrations seem to result in no further increase in FAC rates. EPRI recommends the use of high levels of hydrazine (>100 ppb) to protect steam generator tubes; however, this can result in accelerated rates of FAC in the feed train. Although CHECWORKS does not currently model high hydrazine conditions, any model updates performed after the release of version 1.0F should carefully consider hydrazine concentrations.

   f. Zinc Injection - Industry experience has shown that zinc injection decreases corrosion and FAC wear rates due to the concentration of zinc at the oxide surface. The amount of reduction depends on the amount of zinc at the surface.

ATTACHMENT 9.1 **GUIDANCE ON PARAMETERS AFFECTING FAC**

Sheet 2 of 3

2. Piping Geometry - Piping geometry is one of the most important factors in FAC. Generally, geometries that produce the greatest turbulence also produce the highest FAC rates. Listed below are examples of obvious items that should be considered in any assessment:

   a. Addition or replacement of fittings, bends and branch connections.

   b. Like for like replacement of any fitting in a system that is susceptible to FAC damage or is part of system that is already part of the FAC Program.

   c. Alterations or repairs encountered in the nozzles or walls of FW heaters, MSR, Drain Tanks, FW Pumps, HD Pumps or CD/CB Pumps.

   d. Throttled Valves.

3. Piping Material Composition - Alloying elements improve the resistance of piping systems to FAC. In ascending order of resistance, the following table presents the degree of improvement over carbon steel:

Material	Nominal Composition	Rate (carbon steel) / Rate (alloy)
P11	1.25% Cr, 0.50% Mo	34
P22	2.25% Cr, 1.00% Mo	65
304	18% Cr	>250

4. In-Line Components - Addition or replacement of such components as thermowells, flow elements and pressure-reducing orifices should be evaluated. The local effects caused by these components can generate FAC damage in areas where overall conditions don't indicate the need for inspections.

5. Component Supports - Additions or deletions of components supports which could result in the need for a review of the existing code minimum wall value or a new code minimum wall calculation.

6. Operational Changes - System operational changes such as the normal operation of emergency heater drains, switching of spare components, extended use of normal start-up or by-pass lines, etc.

ATTACHMENT 9.1 **GUIDANCE ON PARAMETERS AFFECTING FAC**

Sheet 3 of 3

7. Component Replacements – Records should be updated for like for like replacement of fittings already in the program including new baseline data, changing next scheduled inspection due date, etc. Note and track whether the replacement components have had surface preparation and a UT grid applied for future outage planning.

8. External Sources – Information concerning FAC Inspection results from other stations and Nuclear Plants operated by others. General information distributed by EPRI Reports, INPO & NRC Bulletins, etc. should also be considered.

9. Maintenance History – A review of the maintenance performed on valves, orifices, steam traps, etc. should be considered. Valves that have had seat leakage can cause very localized wear in systems normally exempted. Plugged traps create water pockets in steam systems that accelerate metal loss. Eroded orifices can cause increased metal loss due to decrease in back pressure and increase in flow rates.

ATTACHMENT 9.2 FLOW ACCELERATED CORROSION PROGRAM ATTRIBUTES

Sheet 1 of 1

PROGRAM ATTRIBUTES

Attributes:

Program Infrastructure

(a) Program Structure: Roles & Responsibilities, Program Ownership, Organizational Interfaces, etc.
(b) Configuration management
(c) Program Bases
(d) Engineering Documentation
(e) Flow Accelerated Corrosion System Susceptibility Evaluation, Latest Revision.
(f) CHECWORKS models
(g) Change processes

Program Staffing and Experience

(a) Background and Expertise.
(b) Qualification and training.
(c) Bench Strength
(d) Time Allotment
(e) Industry Participation

Program Implementation

(a) Work control
(b) Inspections
(c) Maintenance and Repairs
(d) Control of Changes and Deferrals
(e) Review of INPO Operating Experience documents, CHUG operating experience, NRC notices.

Health Monitoring:

(a) System Engineering Health reports.
(b) FAC Quarterly Health Reports.

Effective Assessment:

(a) Perform FAC Self-Assessment on a periodic basis or as defined by applicable procedures.

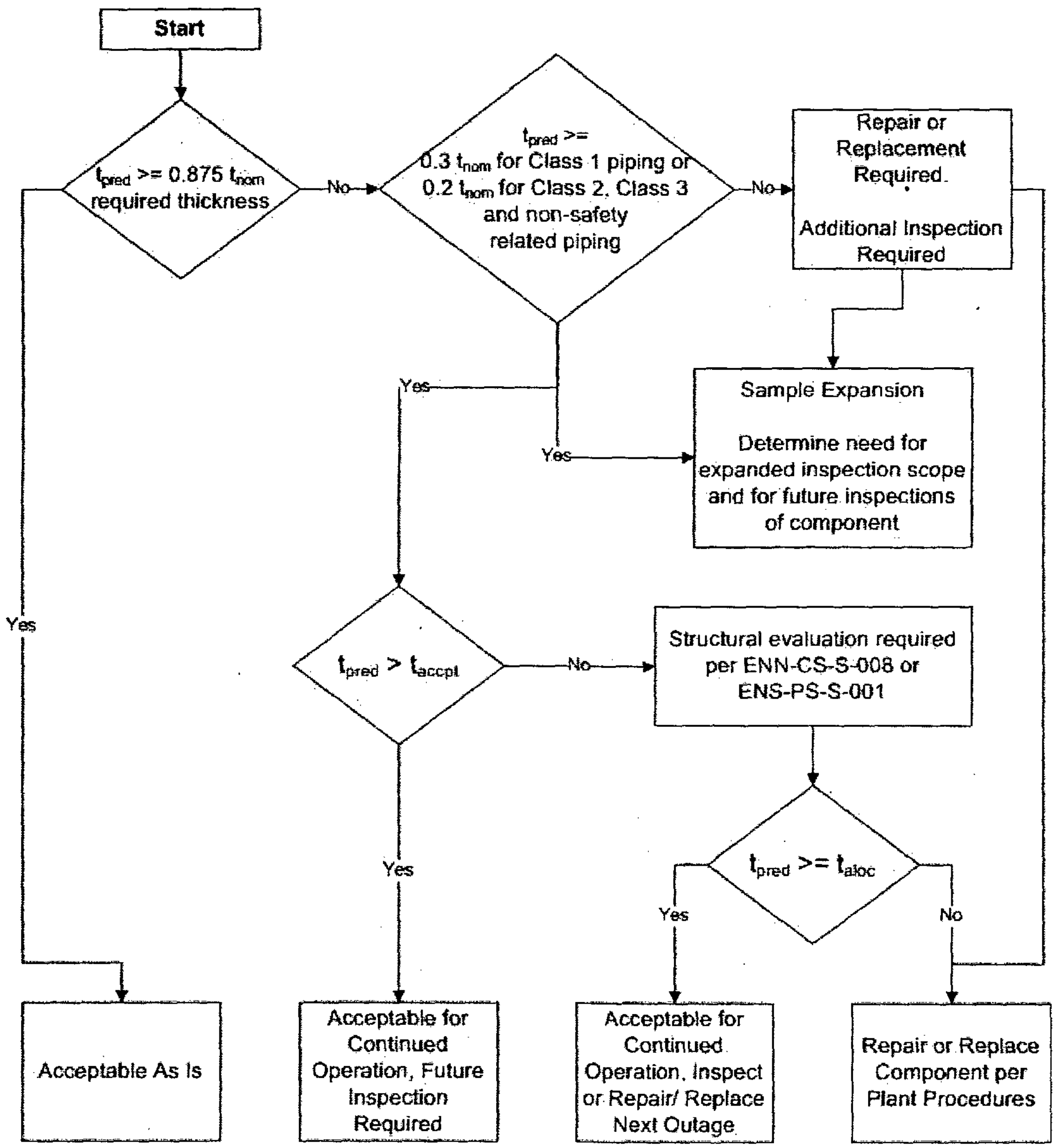
Oversight:

(b) Effective assessment, Benchmarking or Audits.

ATTACHMENT 9.3 **WALL THINNING EVALUATION PROCESS MAP**

Sheet 1 of 1

Logic Diagram - Evaluation of Pipe Wall Thinning



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Indian Point 3
2Q/2007 Plant Inspection Findings

Initiating Events

Significance: Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO INCLUDE THE INTAKE STRUCTURE TRASH RACKS WITHIN THE SCOPE OF THE MAINTENANCE RULE MONITORING PROGRAM
The inspectors identified a Green non-cited violation (NCV) of 10 CFR 50.65(b), in that, Entergy did not include the Indian Point Unit 3 trash rack structures within the scope of the maintenance rule monitoring program. Additionally, Entergy did not demonstrate the performance or condition of the trash racks was being effectively controlled through the performance of appropriate preventive maintenance such that the structure remained capable of performing its intended function. Entergy performed a cleaning of the trash racks to immediately address the lowered service water intake bay level, and they timed service water bay level monitoring to coincide with river low tide cycles. Entergy also entered this issue into the corrective action program as CR-IP3-2007-00453, and developed corrective actions to: modify the requirements for inspection and cleaning of trash racks based on component history and condition monitoring; modify guidance for service water bay level monitoring to be more effective; evaluate maintenance rule system scoping; develop procedural guidance for managing low service water bay levels; and implement a method for monitoring debris fouling of the trash racks.

The inspectors determined that this finding affected the Initiating Events cornerstone and was more than minor because it was similar to Example 7.d in Inspection Manual Chapter (IMC) 0612, Appendix E, "Examples of Minor Issues." Specifically, equipment performance problems were such that Entergy was unable to demonstrate effective control of the performance or condition of the trash racks through appropriate preventive maintenance as specified by 10CFR50.65 (a)(2). The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. (Section 1R12)

Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE PROCEDURE FOR RECIRCULATION SUMP INTERFERENCE REMOVAL
A Green, self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified, in that, Entergy's work package failed to ensure that piping interference was correctly planned for and removed during modifications to the vapor containment and recirculation sumps. On March 9, 2007, during the sump modifications, a section of pipe was cut for interference removal which was different from the piping specified in the work package. This resulted in approximately 385 - 500 gallons of reactor coolant being discharged from the reactor loops into the recirculation sump where personnel were working. The cause of the improper pipe being cut was misidentification of the piping by work planners, followed by a failure of workers to follow steps in the work package that should have identified the work package inadequacy. Immediate corrective actions included a revision to the work package that subsequently welded a cap on the open piping leading from the reactor coolant drain tank to the work site, and plant configuration tags were placed on the residual heat removal interface valves (SI-864E and 864F) to isolate the work area. Entergy entered this issue into the corrective action program as CR-IP3-2007-01059, performed a root cause analysis, and conducted a human performance error review.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, cutting the wrong pipe resulted in the inadvertent draining of reactor coolant system inventory and increased the likelihood of a loss of inventory control. This finding was evaluated using Phase 1 of IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors evaluated the plant conditions (cold shutdown, reactor coolant system open, refueling cavity less than 23 feet) in accordance with Checklist 3 of Appendix G, Attachment 1, and determined that the finding was of very low safety significance because it did not satisfy the criteria of Table 1 for a "Loss of Control," and the Checklist 3 criteria for maintaining adequate mitigation capability (Core Heat Removal Guidelines, Inventory Control Guidelines, Power Availability Guidelines, Containment Control Guidelines, and Reactivity Guidelines) were met.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the work package used for interference removal was not accurate and did not ensure the correct section of piping was identified and appropriately controlled. (Section 1R17)

Inspection Report# : 2007002 *(pdf)*

Significance: Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE PROCEDURE FOR CONDUCT OF RTD CROSS CALIBRATIONS
A Green, self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified, in that, Entergy failed to ensure that appropriate procedures existed to prevent conflicting activities which led to the opening of the pressurizer power operated relief valves (PORVs) when plant conditions did not require them to be open, leading to a partial plant depressurization during plant heat-up. Entergy entered this issue into their corrective action program as CR-IP3-2007-01691. Entergy took immediate corrective action to stop the reactor coolant system pressure transient, and they generated corrective actions to clarify the applicable procedure pre-requisites.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the lack of procedure clarity and poor interpretation of a procedure pre-requisite led to a loss of reactor coolant system pressure as a result of the pressurizer PORV actuation. This finding was evaluated using Phase 1 of IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At Power Situations." The inspectors determined that the finding was of very low safety significance because assuming the worst case degradation, the loss of inventory did not exceed the Technical Specification limit for identified reactor coolant system (RCS) leakage, and the finding would not have caused a total loss of another mitigating system safety function.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the applicable procedure prerequisites were not adequate as written to prevent a plant transient. (Section 1R20)

Inspection Report# : 2007002 *(pdf)*

Mitigating Systems

Significance: Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE PROCEDURE FOR CONTROL OF TEMPORARY MODIFICATION
The inspectors identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Entergy failed to generate a procedure of a type appropriate to the circumstances associated with the implementation of a temporary modification to normal control-room lighting power. The procedure that was generated lacked precautions, limitations, and prerequisites to prevent a low lighting condition in the control room from existing during implementation of the temporary modification. Consequently, during implementation of this temporary modification there were several control panels that did not have adequate lighting for operators to conduct control board manipulations. Entergy entered this issue into the corrective action program as CR-IP3-2007-00821, took immediate corrective action to add additional lighting to the control room, and generated a contingency procedure to allow backup lighting to be energized, if needed.

The inspectors determined that this finding was more than minor because it caused an actual condition to exist in the control room where lighting at selected control panels was not adequate, and contingency plans were not developed for